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PROCEEDINGS OF A SCIENTIFIC-TECHNICAL CONFERENCE

DEDICATED TO THE 20-th ANNIVERSARY OF KOZLODUY NPP WITH INTERNATIONAL PARTICIPATION

OCTOBER 25-26, 1994
KOZLODUY, BULGARIA

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To Prof. Dr. Tsvetan Bontchev, a man of human character and a scientist with appreciation and gratitude

Staff of Kozloduy NPP

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Assoc. Prof. Georgy Glukhov	Technical University, Sofia

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LADIES AND GENTLEMEN,

DEAR GUESTS,

DEAR COLLEAGUES,

The present scientific technical conference THE BULGARIAN NUCLEAR ENERGETICS: STATUS AND PROSPECTS dedicated to the 20^{th} anniversary of the NPP Kozloduy foundation is the first one of this kind in our country. The idea for the conference and its organization belongs to the Plant's administration. Today we can ascertain that the interest to the Plant on the side of Bulgarian experts as well as of experts from other countries is considerably larger than it was expected. That's why let pay no attention to some inconveniences and first of all to the overloaded agenda. It is well known that the most valuable result from the scientific conferences are not so much the meetings as the contacts established between people from different organizations and different countries.

Nuclear energetics establishment and development is something unique in the scientific technical progress history. It has no forerunners like the steam-boilers for instance which starting from the water pump engines in the mines and evolving for decades reached the up-to-date heating plants foundation. The nuclear energetics was born with all its power for number of years on the basis of laboratory experiments as a result of the intellect and efforts of the most talented people of our century: physicians, chemists, engineers specialized in different fields. These people gave to the human race the only practically boundless source of energy.

Of course the rapid growth of the nuclear energetics required the most attention in relation to deportment of the metals subject to a powerful neutron radiation, the possibilities for automation and a series of other problems. Some now concepts originated for increase of the environment protection. This led to the sealed areas creation. An extremely important part for the continuous improvement of the nuclear plants played the development of electronics, the powerful computers creation, the invention of new types of nuclear radiation detectors, improvement and creation of new control systems based on the ultrasonics, eddy current, magnetic methods and many others. Briefly, the nuclear energetics would not be able to develop without the current scientific achievements as in a number of cases it stimulated them itself. That's why a conference like the present one seems to be something quite natural. Of course in the large countries, especially these which founded and developed the building of reactors there are huge scientific centers and this kind of activity has different dimensions.

The Bulgarian nuclear energetics existence without accidents within twenty years indicates that NPP Kozloduy has at its disposal highly qualified engineering staff. In conjunction with that we can claim that our country is in possession of powerful scientific and engineering potential in almost all fields relating to the nuclear energetics. Unfortunately through reasons we have no chance to discuss now our science contributed quite slightly for the plant growth. That's why today we have to appreciate the initiative of the plant management to organize the present conference.

At present NPP Kozloduy produces over 40% of the power supply needed to the country with this production caused no damage to the environment. It is a duty for all of us not only to assist in the range of our capabilities for the plant development but to dispel the accumulated against it prejudices, even more accusations originating from mercenary motives. Let's hope that the present conference will play its part with respect to this.

I take the opportunity to express the Bulgarian participants' gratitude to our colleagues from other countries who showed interest and found the time to take part in the conference. Let's hope this will help us to establish even better cooperation.

On behalf of the organization committee I wish all of you beneficial participation in the meetings, discussions and establishment of enduring links between us.

President of the Organization Committee

Prof. Dr. Tsvetan Bonchev

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АЭС "КОЗЛОДУЙ" - ПЛЮСЫ И МИНУСЫ

К. Кузманов - Генеральный Директор АЭС „Козлодуй"

Уважаемые участники конференции,

Уважаемые гости,

Настоящее сообщение не имеет характера научного сообщения. Это просто попытка поделиться с вами мыслями об атомной энергетике, о ее применении в Болгарии и об АЭС „Козлодуй" по поводу ее 20-летия.

Всем известно, что после трагедии в Хиросиме и Нагасаки в конце второй мировой войны создатели атомной бомбы и их коллеги во всем мире поняли, что создано оружие массового уничтожения. Этот факт спровоцировал разум политиков и ученых, и реакция мировой общественности не заставила себя ждать! Родились два знаменитых призыва: "Перекуем мечи на орала!" (They Shall Beat Their Swords into Plowshairs)" и другой "Атом для мирных целей! (Atoms for Peace)", которыми и сегодня руководствуется человечество в его борьбе за всеобщий прогресс. Не отрывая внимания от военных интересов, большая часть фундаментальных научных исследований были переориентированы на более универсальное применение не только в промишленности, но и в материальной сфере общественной жизни в целом.

В ответ на это в 1951 г. на полигоне Арко в штате Айдахо, США, были произведены первые киловатт-часы электроэнергии экспериментальным реактором на быстрых нейтронах EBRD1 (Experimental Breader Reactor) мощностью 100 КВт. В 1954 г. его примеру последовала первая в мире 5- мегаватовая Советская АЭС промышленного значения в г. Обнинске, затем в 1956 г. - французский тяжеловодный реактор G1 в Маркуле мощностью 40 МВт и газографитовый реактор в Калдер Холле (Calder Hall) мощностью 50 МВт в Великобритании. Очевидно, ученые развитых стран мира отдавали свой труд для создания подходящей базы для целесообразного использования ядерной энергии в мирных целях. В результате этого стало возможным проведение Женевской конференции, которая через 10 лет после трагедии в Хиросиме и Нагасаки под девизом "Атом для мирных целей", послужила началом новой эры в развитии человечества - эры превращения ядерной энергетики в жизненный фактор не только для великих сил, но и для маленьких государств. В эти годы после войны начинает формироваться государственное регулирование при использовании ядерных источников посредством принятия и введения в действие первых ядерных законов.

Первые шаги ядерной энергетики совпали с бурным ростом населения на планете и порожденным в результате этого интенсивным развитием энергетики во всех ее тогдашних формах. Это привело к неожиданным отрицательным деформациям в окружающей среде и до тревожного уменьшения высококачественных, но невосстановимых энергетических ресурсов. Заканчивалась эпоха нефти - дешевого и пригодного для всех потребителей энергоисточника, и нужно было искать новые, более эффективные пути для борьбы с энергетическим дефицитом. Эти пути нашли свое развитие в двух основных направлениях.

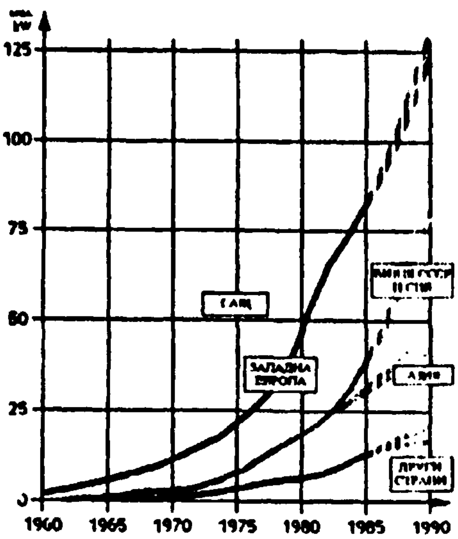
1. Экономия энергии во всех областях материальной сферы посредством последовательной, твердой и хорошо управляемой ценовой политики. Были созданы условия для прекращения беспорядочной траты энергии и задержки роста потребления энергии на продолжительный период времени. Были созданы новые низкоэнергопоглощающие технологии и новые прогрессивные энергоизточники. Это стремление породило и второе направление в энергетике.
2. Замещение нефти более дешевыми .1 надежными энергоносителями.

Первое место в этой замене приходилось на природный газ, который благодаря незначительным необходимым капиталовложениям и удобству при его использовании предлагал хорошие перспективы для удовлетворения нужд населения в энергии.

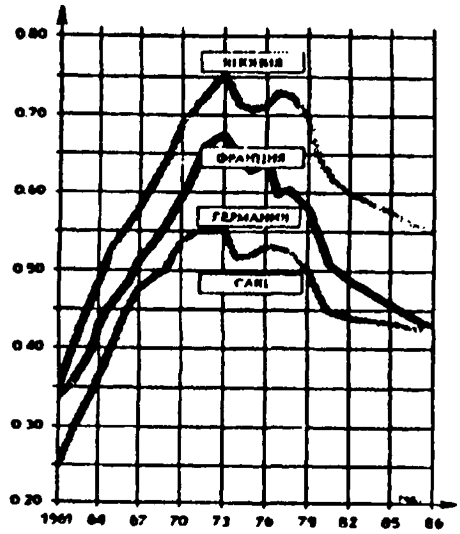
На втором месте стояла возможность увеличения доли угля в общем балансе энергетики. Трудности по его транспортировке, а также значительный выброс серных, азотных и углеродных окисей, выделяемых при его сгорании, затруднили повсеместное их использование. В перспективе они могли быть использованы главным образом при решении местных энергетических проблем (например, в Польше и Германии, в комплексе "Марица-Изток" в Болгарии).

В результате интенсивной научной и инженерной деятельности в середине шестидесятых годов в промышленно развитых странах ядерная энергетика уже доказала свои возможности для дешевого и экологически чистого производства, независимо от политической, социальной, психологической и административной конъюнктуры.

Потребление нефти значительно уменьшилось (рис. 1), а после 1970 г. доля АЭС в мире значительно выросла (рис. 2). Во многих страна мира ясно вырисовывалась тенденция принять вызов одного великого открытия и поставить его на свою службу. Среди них была и наша маленькая Болгария, которая з конце 60-х выбрала ядерное электропроизводство в качестве энергетической перспективы. Началось строительство первой на Балканах АЭС на реке Дунай около города Козлодуй. Это позволило нашей стране занять место в ряду стран-пионеров, имеющих атомные станции, и с этого момента она достойно занимает это место.



Фиг. 1 Количество нефти, потребляемой для получения первичной энергии.



Фиг. 2 Ядерные мощности, установленные в мирэ за период 1960 - 1990 г.

Выиграли ли мы или проиграли от АЭС "Козлодуй"? Каков итог работы станции для Болгарии и болгарского народа?

Давайте сначала рассмотрим, что мы потеряли, что теряем и что можем потерять.

1. Безвозвратно потеряны около 200 га плодородных земель около села Хырлец и г. Козлодуй, необходимых для строительства станции и вспомогательных хозяйств около нее.

2. За короткий срок израсходованы крупные для страны капиталовложения (включительно и для сооружений, обеспечивающих нормальную работу энергетической системы при наличии нескольких 1000 мегаваттовых электрогенерирующих мощностей). Это, конечно же, сказалось в определенный период на общем экономическом состоянии страны. Однако, этот компромисс был необходим, потому что энергетическая обстановка в стране требовала собственных энергоисточников, а единственной альтернативой был восточно-маришкий лигнитный бассейн. Но для строительства одной ТЭС на этой базе нужны были соответствующие капиталовложения, т.к. кроме самой станции, были необходимы обширные площади для угольных складов и шлакохранилищ, сложные и тяжелые машины и сооружения для предварительной переработки угля, для сооружения самих рудников.

3. При более тяжелой радиационной аварии АЭС создает опасность радиоактивного заражения окружающей среды (включительно и в соседней

Румынии) и радиационного облучения работающих на АЭС и людей, живущих в непосредственной близости от станции и всех многолетних последствий этого.

Может быть, здесь уместно остановиться более подробно на этой возможности. В самом деле, очень мало больших промышленных объектов, которые вызывают такую сильную оппозицию и так много вопросов, как атомные станции. Неосновательных страхов от несовершенства производственных процессов с тяжелыми воздействиями на окружающую среду намного больше, чем истинных результатов от вредных выбросов и неизбежных помех, сопровождающих производство. Именно поэтому мы, работающие в АЭС "Козлодуй", как и все связанные с производством специалисты и ученые, должны показать обществу ясные и точные документы, свидетельствующие о безопасности и защите от йонизирующей радиации, о воздействии на окружающую среду, а также и принятые меры по ее сохранению.

Все присутствующие знают, а и общественность должна знать, что ядерный взрыв в реакторе физически невозможен, но радиоактивность отходов, накопленных при нормальной эксплуатации, представляет потенциальную опасность при выбросе этих продуктов из реактора. Два особых случая, которые произошли при различных условиях, напоминают нам об этом всегда! Неисправность в Три Майл Айланд в 1976 г. вывела ее из строя, но авария, ограниченная герметическим помещением реактора, не нанесла вреда окружающей среде. В Чернобыле, однако, в 1986 г. было не так. Отсутствие плотной преграды позволило продуктам ядерного деления распространиться вне станции, но эта авария, вопреки катастрофальным последствиям, показала максимальное зло, которое может случиться при эксплуатации атомной станции. И сейчас, когда эмоции уже в прошлом, нужно объяснить всем, что риск при ядерном электропроизводстве меньше, чем на предприятиях тяжелой промышленности. Вызванное причинами, которые едва могут появиться в водородных реакторах, а тем более в реакторах, отвечающих современным требованиям, это событие не смогло остановить атомные станции, но помешало ядерным государствам, включительно и Болгарии, выполнить свои программы по развитию атомной энергетики. К сожалению, психологические осложнения и в будущем будут мешать рассматривать ядерную промышленность как любую другую. Поэтому, кроме обязательств о безопасной эксплуатации атомной станции, перед нами как операторами стоит задача сделать как можно больше, чтобы показать, как мы защищаем человека и природу. Необходимо объяснять, какие меры безопасности мы принимаем, как предохраняем общество строгим соблюдением норм радиоактивных отходов и способов их хранения на станции, постоянным контролем, радиационной защитой и всех параметров, связанных с охраной окружающей среды.

4. Станция производит радиоактивные отходы, включительно и отработанное ядерное топливо, которое сохраняется на ее территории - это усложняет

эксплуатацию и создает дополнительные проблемы, для решения которых необходимы значительные капиталовложения (переработка радиоактивных отходов в удобные для сохранения формы - такая установка строится в АЭС "Козлодуй" по технологии американской фирмы "Уестингхаус", которая должна войти в строй в начале 1996 г.; склад для сохранения отработанного топлива; национальный склад для радиоактивных отходов).

5. Необходимо отчислять значительные финансовые средства для вывода станции из эксплуатации после исчерпывания ресурса ее основных сооружений (минимум 5% в год от общих капиталовложений).

6. Необходимо непрерывно повышать уровень безопасности и эксплуатации посредством реконструкции и модернизации, которые, в свою очередь, требуют постоянного выделения средств и стабильного финансового обеспечения этой безопасности.

7. Отсутствует возможность гибкого управления электрической нагрузкой, что создает серьезные затруднения в управлении энергетической системой, особенно в случае минимальных нагрузок.

8. Необходима постоянная подготовка персонала, которая требует дорогих средств обучения: тренажеры, макеты, системы обучения.

Верятно, здесь перечислены далеко не все минусы АЭС "Козлодуй", но это основные, около которых можно сгруппировать и другие минусы, которые, вероятно, здесь не упомянуты. У нас есть возможность их обсудить на данной конференции.

А сейчас давайте противопоставим пользу от АЭС "Козлодуй" и ее преимущества перед остальными источниками энергии в Болгарии.

1. Станция - это независимый энергоисточник. Ее работа не зависит от атмосферных условий, от социальной и политической обстановки в стране, в известной мере и от международной обстановки.

2. Она является надежным источником с высоким коэффициентом использования. С момента первого включения в параллель с энергосистемой в июле 1974 г. до сегодняшнего дня произведены более 200 миллиардов киловатт-часов электроэнергии, а сейчас, после ввода в эксплуатацию второго тысячника, АЭС "Козлодуй" - это основной источник энергии страны, и она удовлетворяет около 40% нужд в электроэнергии.

3. Производимая электроэнергия имеет низкую себестоимость и в два-три раза дешевле электроэнергии, производимой теплоэлектростанциями. Даже если учесть капиталовложения для сохранения радиоактивных отходов и расходы на

вывод из эксплуатации, то стоимость электропроизводства продолжит следовать мировым тенденциям более низких, чем от других энергоисточников, разходов.

4. АЭС "Козлодуй" не выбрасывает вредные отходы. Благодаря этому сэкономлен выброс десятков миллионов тонн пыли через трубы и шлакохранилища на теплоэлектростанциях, а также миллионы тонн вредных окисей. В этом отношении у станции нет другого подобного экологически чистого конкурента. И еще один факт. Я не против ТЭС, потому что без них с трудом могут быть решены энергопроблемы страны. Хочу отметить, что радиационный фон около ТЭС значительно выше, чем около АЭС "Козлодуй".

5. АЭС "Козлодуй" дала сильный толчок для развития нашей науки, инженерно-конструкторской и проектной деятельности, дала стимул для появления новых отраслей промышленности, развития учебного дела, медицины. Она помогла открытию новых рабочих мест, развитию города Козлодуй и близлежащих сел. Косвенно она положительно повлияла на развитие культуры и общего социального уровня населения.

6. АЭС "Козлодуй" создала мощную группу ядерных специалистов, которые не только успешно эксплуатируют станцию, но и интенсивно работают для ее развития. Выдающиеся ядерные специалисты не раз давали о них лестные отзывы.

7. АЭС "Козлодуй" создает и поддерживает положительное мнение о Болгарии в других странах и этим помогает стране не только производить электроэнергию, но и создает завидную репутацию тем, что занимает значимое место в ряду мирных ядерных мощностей мира.

Этой короткой оценкой АЭС "Козлодуй" мы определили ее плюсы и минусы. Если их сравнить, то увидим, как ярко выделяется большая доля станции в экономике Болгарии, в обеспечении нормальной и чистой окружающей среды. Все мы, работающие в АЭС Козлодуй и неразрывно связанные с ней, никогда не должны забывать, что она будет жизненной и полезной только тогда, когда мы всеми силами будем заботиться о ее безопасности.

г. Козлодуй
25.10.1994 г.

К. Кузманов

BG9600393

CONFERENCE SCIENTIFIQUE ET TECHNIQUE A L'OCCASION DU VINGTIEME ANNIVERSAIRE DE LA CENTRALE NUCLEAIRE DE KOZLODUY LES 25 ET 26 OCTOBRE 1994 KOZLODUY - BULGARIE

Intervention de Monsieur Rémy CARLE - Président de WANO

Mesdames, Monsieurs, bonjour,

Cher Monsieur KOUZMANOV, je vous remercie d'avoir invité le Président de WANO à cet anniversaire et de me donner l'occasion d'exprimer quelques réflexions, ce matin, à l'ouverture de cette conférence. 20 ans, cela représente beaucoup de KWh fournis à ce pays, mais aussi, beaucoup de dévoument et de compétences mis au service d'un grand projet.

Et je voudrais d'abord vous en féliciter,vous personnellement, et tous ceux qui ont contribué.

Sans doute ces vingt ans n'ont - ils pas coulé comme coule le grand Danube tranquille qui refroidit la centrale.

Pendant ces vingt ans, l'énergie nucléaire a concidérablement évolué et un réacteur conçu autour de 1970 a bien mal, dans tous les pays, à obéir aux règles de sureté de 1994.
Et puis surtout, ce pays a connu au cours de ces vingt années une histoire difficile sur laquelle je n'ai pas besoin d'insister car vous la connaissez mieux que moi.

Le résultat de tous ces événements a été d'abord, à mon sens, l'isolement de votre centrale. Si des journalistes occidentaux ont pu qualifier Kozloduy de centrale la plus dangeurese du monde, eux qui ne savent pas distinguer un VVER d'un RBMK, c'est d'abord à cause de cet isolement, du sentiment d'une centrale mal connue, dans un pays lointain, sans connexion avec le reste du monde. C'est cet isolement que WANO s'est attaché à rompre et telle est bien sa mission primaire.

Isolement vis - à - vis de l'ouest, et le jumelage de Kozloduy avec la centrale de Bugey en France est de ce point de vue exemplaire.

Isolement aussi en Europe de l'est: nous devons toujours nous souvenir que le VVER ont été conçu, réalisés et exploités par tout un réseau de compétences, d'Instituts, d'usines de fabrication mis en place sur l'ensemble de l'ex - Comecon et ce tissu ne pouvait etre impunément déchiré.

Isolement entre exploitants, mais aussi entre autorités de sureté; le role de l'AIEA mais aussi la collaboration entre BNSA et Riskaudit sont des éléments essentiels de l'insertion de Kozloduy dans la communauté nucléaire internationale.

Cette conférence meme par le nombre et la diversité de ses participants, par le nombre de papiers qui y seront présentés, démontre le réseau d'intérets et de collaboration qui sont aujourd'hui bien vivants autour de votre centrale et de cela aussi il faut vous féliciter.

La sureté, nous le savons tous, est le produit de la qualité des équipements par la qualité de l'exploitation.

Bien entendu certaines insuffisances d'équipement sont incontournables, je sais que vous etes attentifs au d'améliorer le combustible, le controle - commande, l'amélioration électrique et bien d'autres choses encore dont vous parlerez pendant cette conférence. Mais il est claire qu'au delà du matériel, la qualité, la rigueur de l'exploitation sont essentielles. Et elles doivent etre pensées précisément en fonction de l'installation telle qu'elle est avec ses qualités et ses déficiences. Ceci est une question de management, de discipline, de comportement quotidien, d'acceptation du controle d'autrui, en définitive, comme l'on dit aujourd'hui de culture de sureté. Et ceci à tous les niveaux. Vous en parlerez certainement pendant ces deux jours pour dire, comme l'a dit Haus BLIX, quand il est venu ici, il y a quelques semaines, que beaucoup a été fait mais que beaucoup reste à faire.

Je me souviens lors de ma première visite avoir demandé à voir les procédures d'exploitation. Et on a sorti de l'armoire de beaux cahiers bien reliés mais à l'intérieur l'encre s'était effacée et rien de tout cela ne pouvait servir. Aujourd'hui cela a été corrigé,mais croyez le bien, l'encre s'efface toujours avec le temps, au moins dans les esprits.

Que ce soit en Bulgarie ou en France, ou aux Etats - Unis ou en Russie, la culture de sureté est à refaire tous les jours. WANO est bien décidée à continuer à travailler avec vous en ce sens mais bien sur nul ne peut le faire à votre place.

Et c'est bien la condition pour que vous puissiez,en toute sérénité,répondre aux critiques que Kozloduy peut continuer à fonctionner comme n'importe quelle autre centrale du parc nucléaire mondial.

Je sais les difficultés qui sont les votres pour améliorer matériels et personnel jour après jour,avec des moyens qui sont très limités.

Vous avez besoin pour cela de l'appui de votre gouvernement qui fait par ailleurs face à de multiples problèmes.

Vous devez le convaincre que non seulement Kozloduy est tout à fait esentiel pour l'alimentation électrique de la Bulgarie mais qu'il constitue le premier pas d'un développement nucléaire de longue durée.

Kozloduy n'est pas là par hasard; non Kozloduy n'est pas un accident de l'histoire dont il faudrait simplemet profiter à court terme.

La Bulgarie ne dispose que de ressources énergétiques conventionnelles limitées et de médiocre qualité, seule l'énergie nucléaire pourra lui éviter des importations massives et couteuses lorsque la vie économique du pays reprendra son essor.

Par ailleurs, il est claire que cette région du sud est de l'Europe au confluent de l'Europe de l'ouest, du monde ukrainien et russe et du Moyen - Orient occupe une position de première importance dans les échanges d'électricité de domain. Et vous devez réfléchir me semble-t-il, `. y prendre votre place, peut-etre en association avec certains de vos voisins -j'étais la semaine dernière à Cernavoda - grace à l'outil puissant de l'énergie nucléaire.

Développer en ce pays des compétences nucléaires,. une qualité nucléaire, une culture nucléaire n'est pas un investissement à court terme mais bien au contraire une ouverture sur l'avenir.

N'est -ce pas un grand encouragement,un grand espoir,notammentpour les jeunes qui cherchent les voies du meilleur service de leur pays? N'est -ce pas une raison majeure pour le gouvernement de ce pays de donner à l'amélioration de Kozloduy la priorité nécessaire?

Introduire Kozloduy dans la communauté internationale.

Amener et maintenir son niveau de sureté au niveau international.

Préprer le nucléaire de domain.

Tels sont les objectifs auxquels vous vous consacrez.

WANO continuera à vous y aider.

Cette conférence y contribuera certainement,je lui souhaite le meilleur succès.

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SAFETY OF NPP WITH WWER-440 AND WWER-1000 REACTORS

E. Balabanov - Energoproekt Plc,
J. Gledatchev, D. Angelov, Kozloduy NPP

The objective of this paper is neither to evaluate the safety of WWER-reactors, nor to impose any requirements on the power plant. It just points out some facts and problems which had to be accounted for during the last years of operation of Kozloduy NPP.

At present in the world 420 nuclear units produce 17-18% of the electricity and 77 more units are under construction. The percentage of the nuclear energy in different countries varies in a very wide range. For example in France Nuclear power plants produce 75% of the electricity, in Belgium - 60%, in Hungary, Finland and Korea - 50%, in Bulgaria - approximately 35%, In former Chechoslovakia - about 30% and over 12% in the countries of the former USSR. It should be noted that most of the units in Western Europe are in operation, while in Eastern Europe are most of the units under construction. (Figs. 1, 2). The reactors designed in the former USSR are 15% of all reactors in operation and 42% of those under construction. Of all soviet designed reactors the majority are pressurised water reactors (WWER) and at present 44 of these are in operation and 25 under construction. The distribution of the reactors in the different regions of the world with respect to their age is given in Fig. 3.

At present 10 reactors WWER-440/230, 16 reactors WWER-440/213 and 18 reactors WWER-1000 of different modifications are in operation. These operated in Bulgaria are a considerable part of them (Fig. 4).

This statistics shows that 14% of all WWER reactor in operation are at Kozloduy NPP. The total power of Kozloduy NPP is 9% of all nuclear power installed in the eastern part of Europe.

WWER-440

The WWER-440/230 units are designed during the sixties, with the main objective of maximum electrical output and maximum availability. This is reflected in the design basis and later on in the technical solutions such as: low power density in the core; three levels of reactor control and protection before a reactor scram; six primary loops with primary isolation valves both in the hot and in the cold legs; horizontal steam generators with a relatively large water inventory; two turbines, allowing operation at different power levels; a large number of cross-unit connections. allowing common use of systems of neighbouring units.

On the other hand, the big safety margins of most of the technological parameters permit the successful restoration from many transients

The design basis of these unit allow maintenance as well as some repairs „on-line", which is also helped by the low radiation level of the units and the number of inter-unit connections. The applied design solutions, providing a convenient access to the equipment during operation and the operational practice giving a priority to a flexible mode of operation, result in relatively high availability of the units.

The design of the Kozloduy WWER-440 units is following the regulations of the former USSR during the sixties and seventies. These follow mainly a deterministic approach, so the WWER-440/230 units differ from the contemporary safety requirements, some of them not being fulfilled.

For this reason the safety concept of WWER-440/230 applied for its design is different from the modern safety concepts. Some of the major deviations are:

- A low level of the design basis accident;
- Insufficient configuration of the emergency core cooling systems to prevent core damage in case of a complete rupture of a primary loop;
- Common cause failures are not considered;
- Insufficiency of the last physical barrier to prevent radiological releases to the environment;
- Incomplete design solutions with respect to external events;
- Lack of a complete set of design solutions to minimise/prevent the consequences of human errors;

On the other hand the design concept of WWER-440/230, aimed at maximum electrical output, provides some specific characteristics of the plants, enhancing their safety.

These characteristics give the plants features of „internal, incorporated' safety, which is now becoming a part of the design of the plants of the new generation.

These safety features of the WWER-440/230 design are not typical for the new commercial designs of pressurised water reactors. As typical safety features of this type the following can be mentioned:

- Low thermal density in the core and, respectively, a compact reactor core with excellent neutron-physical characteristics with respect to the Xe oscillations. This makes the control of the power shape easy during transient modes and eliminated the demand of special control equipment for suppression of Xe-oscillations;
- Low specific thermal loading of the fuel rods and, as a result, a relatively low heat flux and large DNBR values for a large spectrum of transients. The average fuel temperature during normal operation is relatively low and the gaseous fission products are better contained within the fuel matrix (Fig. 5)[2];
- A large coolant inventory in the primary circuit and in the steam generators. This characteristics is unique for the WWER-440 type of reactors. The large thermal capacity of the reactor coolant systems provides a natural protection in cases of disbalance between heat generation and heat removal. This feature plays a positive role during accidents and transients as follows:
  - The large thermal inertia of the NSSS makes the plant less sensitive to a lot of operational deviations. In most of the PWR types of reactors operational transients initiated in the secondary circuit (and having a higher frequency), lead to more severe transients in the reactor coolant system, often requiring the operation of the pressurizer safety valves and increasing the probability of severe accidents with respect to the core behaviour. In the WWER -440 reactors most of the transients initiated in the secondary circuit are

suppressed in the steam generators by a small deviation of the coolant level, while the influence on the primary circuit parameters is rather weak;

- The large primary coolant inventory makes the plants steady against different accidents such as blackout, complete loss of feed water (e.g. due to a fire in the turbine hall). In these cases severe damages of the core can be expected in 4-6 hours, providing sufficient time for appropriate recovery actions (Fig. 6) [2];
- The large primary coolant inventory is an advantage in case of small LOCA with loss of the high pressure injection system. This advantage is a result of the sufficient time for manual depressurization of the primary circuit and long term cooling using the low pressure injection system;
- In cases of anticipated transients without scram (ATWS) the resulting pressure peak is considerably lower than for PWR reactors. In addition to this, the large DNBR decreases the probability of a boiling crisis (Fig. 7) [2];

Except for the in-built safety features, some of the design solutions of WWER-440 benefit their safety characteristics. Such are:

- Multy-loop configuration of the primary circuit. The six primary loops decrease the impact of main equipment failures (such as tripping of a reactor coolant pump, etc.);

- Horizontal steam generators: These enhance the transition to single-phase natural circulation of the primary coolant. It also provides several different routes for decay heat removal both from a tight and from an open primary circuit;

- Primary loops isolation valves on both the hot and the cold legs of each loop provide a possibility for some maintenance operations in the steam generators without draining of the whole primary circuit (which is usually a requirement for the typical PWRs). This eliminates the possibility of a loss of the ultimate heat sink;

- Two turbine-generators. This has a positive effect on the safety parameters of the plant, softening the loads due to transients initiated in the secondary side. The two feed water tanks provides better means for mitigation of transients and accidents. The two independent connections to the electrical grid provides a higher reliability of the electrical supply system of the plant.

Independently of all those positive features, the low DBA level still remains a concern for these units due to the design concepts and their possible upgrades with the objective to bring them closer to the modern safety requirements.

The major deficiencies of the WWER-440/230 can be summarised as follows:

- Insufficient capabilities for emergency core cooling;
- Insufficient diversification and physical separation of the safety systems;
- Deviations from the modern concepts in the control systems;
- Insufficient fire protection capabilities;
- Incompleteness of the last physical barrier (lack of a full containment).

Coming back to Fig. 3, it is obvious that the operation of old plants is not a Bulgarian problem. This is a reason for the development of a world practice to operate nuclear power plants, started under different conditions (old design solutions and old regulations).

Our and the world practice outline the main trends for future upgrades of the WWER-440/230 plants:

- Permanent efforts and short-term measures for increasing of the safety level by improvement of the equipment, operational documentation and safety culture;
- Development of short-term programmes for upgrades, optimising the costs and the expected results. At the same time, for a certain period of time, a special mode of operation may be applied together with compensating measures to improve the safety level;
- Development of long-term strategies, taking into account the remaining resources of the equipment and accurate determining of the remaining life-time.

The main objective of all these programmes should be the preserving and improvement of all the existing positive safety features.

For the last years of operation this approach is already a practice in the operation and management of Kozloduy NPP. The first programmes for safety upgrades of Units 1-4 after 1991 are already completed. They included measures aimed at the improvements of the design, operation, maintenance and safety culture.

The measures completed until this moment fall in the following main groups:

- **Design solutions**
  - Improvements of the integrity of the primary circuit (annealing of the reactor pressure vessel, study of the possibility of the application of the LBB concept, evaluation of the remaining resource of the equipment, qualification of the pressurizer safety valves, avoiding of a pressurised thermal shock);
  - Containing of the radioactive release in the hermetic zone (tightening, local and global leak tightness tests, qualification and modernisation of the containment spray system);
  - Improvement of the I&C systems (qualification of sensors and measurements chains, main control room, information support);
  - Improvement of the electrical supply systems;
  - Improvement of the fire protection (evaluation of the fire hazard, fire detection and announcement systems, fire protection and extinguishing measures);
  - Accident analyses;
  - Reassessment of the seismic response of the plant.
- **Improvement of the operation**
  - Introducing of a new style of plant management;
  - Improvement of the maintenance and repair of the equipment;

- Upgrading and improvement of operational documents, including development and implementation of a new system of normal and emergency operating procedures;

- Implementation of a new system for training of the personnel;

- A new organization of emergency preparedness and accident management;

- Improvement of the safety culture.

Obviously the above listed topics are in compliance with the internationally accepted standards and practices, which inevitably reflects on the current status of the plant.

WWER-1000 UNITS

The nuclear powêr plants with WWER-1000 type of reactors are the third commercial generation of the Soviet design of pressurised water reactors. Historically several modifications of this type of reactors were put into operation:

- Model 187 - Unit 5 of Novovoronezh NPP;
- Model 302 - Unit 1 of South-Ukranian NPP;
- Model 338 - Units 1 and 2 of Kalininskaya NPP;
- Model 320 - Unified for all other plants in operation;
- Model 392 - Unit 6 of Novovoronezh NPP.

The Model 320 is the one which is at present the most popular, having unified equipment and a high enough level of safety.

The Units with WWER-1000/320 are designed following the regulations of OPB-82. For this reason they are in compliance with the safety standards, currently applied in the international practice. The design principles and the design architecture of the safety systems are in complete agreement with the current international practice. As a result the WWER-1000 units are efficient and economically optimised, with efficient safety systems, satisfying the international requirements.

A general comparison of WWER-1000 to similar PWR plants designed and put into operation in the industrially developed countries (USA, France, Germany) show practically identical layout, techological and safety characteristics.

The layout of the plant with a single reactor and turbine and four primary loops provides similar characteristics for the compared plants. The horizontal steam generators with a relatively high secondary coolant inventory and relatively stable SG level, makes the WWER-1000 plants proof against a number of disturbances. On the other hand, the relatively high coolant inventory to reactor power ratio provides considerably better possibilities of prevention of severe core damages under some emergency situations (complete blackout, anticipated transients without scram, etc.).

Table 1

	Coolant inventory/thermal power, m^3/GWth	
	Kozloduy	Konvoy-1300
Primary circuit	108	99
Secondary circuit	103	61

The comparison of the parameters of the fuel rods and reactor core as well of the technological parameters of the primary circuit for the two types of reactors show that their safety margins are very similar.

Besides the facts mentioned above, it should be noted that at present a very extensive experience has been gained in Bulgaria from the start-up and several years of power operation of Units 5 and 6 of KNPP. At the same time regular activities have been performed, such as:

- Reviews and evaluation of the design;
- Analyses of the forced technical solutions during the construction and beginning of operation of the units;
- Analyses of equipment failures and the related quality of the equipment;
- Analysis of the deviations of Units 5 and 6 from the current regulations in the country (BNSA and its orders);
- Analyses related to replacement of equipment, modifications of systems and constructions;
- Safety analyses, etc.

Based on all stated here, the main conclusion can be drawn that as a whole the Units 5 and 6 at KNPP are designed and constructed with respect to the internationally accepted principles of safety, but they need some improvements, related to modernisation of separate systems and equipment, to introducing of some new systems and to improvement of the system of emergency operating procedures. These improvements will bring the units to a level, comparable to the practice in the industrial countries.

On the other hand it should be clear that a high safety level is not just a single effort, but a continuous process throughout the plant operation, with periodical introduction of new principle solutions, so that the safety level would not stay strongly behind the constantly increasing requirements.

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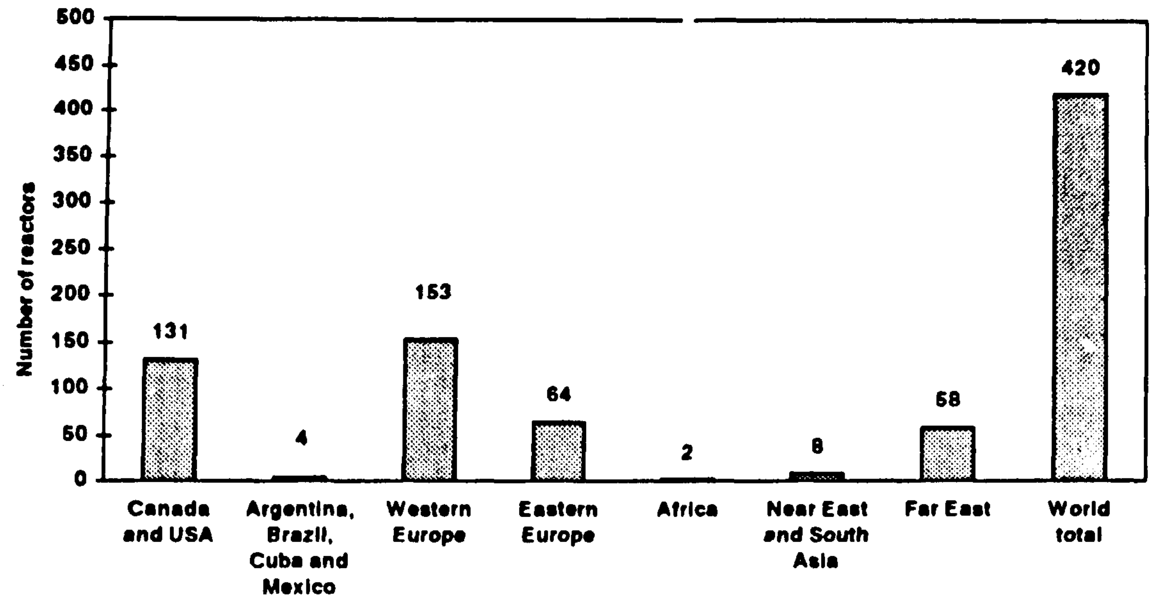


FIG.1 Number of commercial reactors in the world, including 6 reactors in Taiwan, China (December 1991)
Reference: IAEA - Power Reactor Information System (PRIS). Non-government information

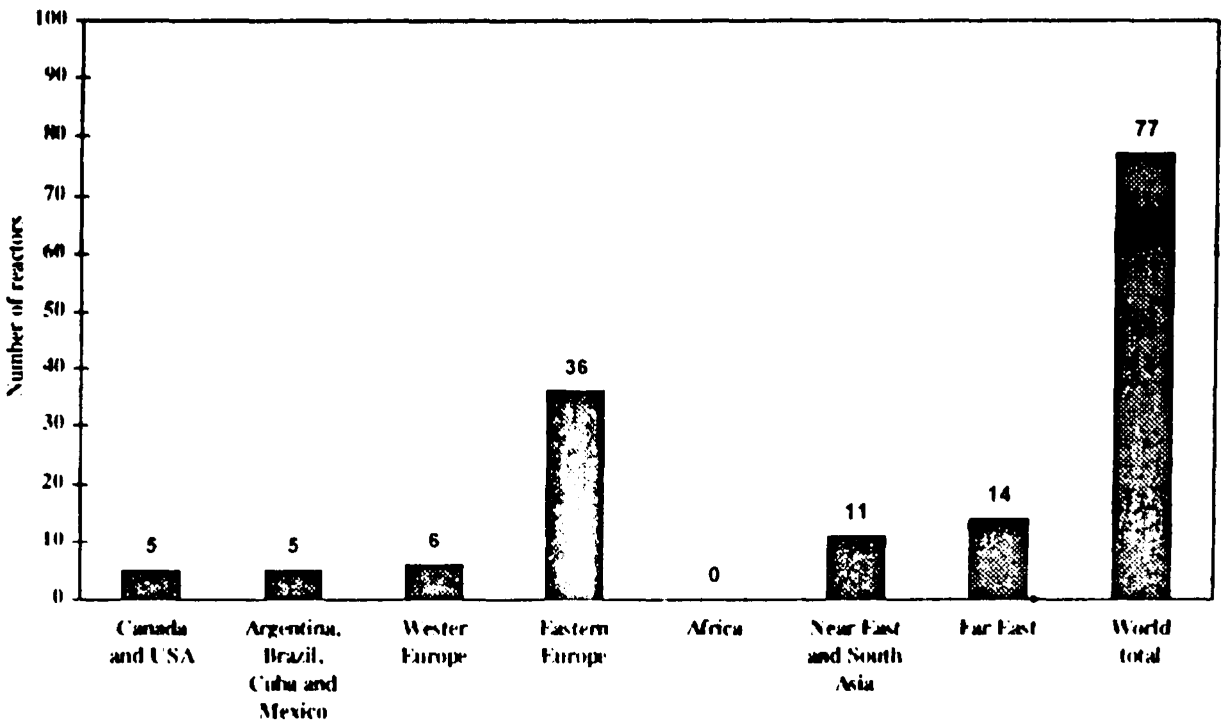


FIG.2 Reactors under construction in the world (December 1991)
Reference: IAEA - Power Reactor Information System (PRIS). Non-governmental information

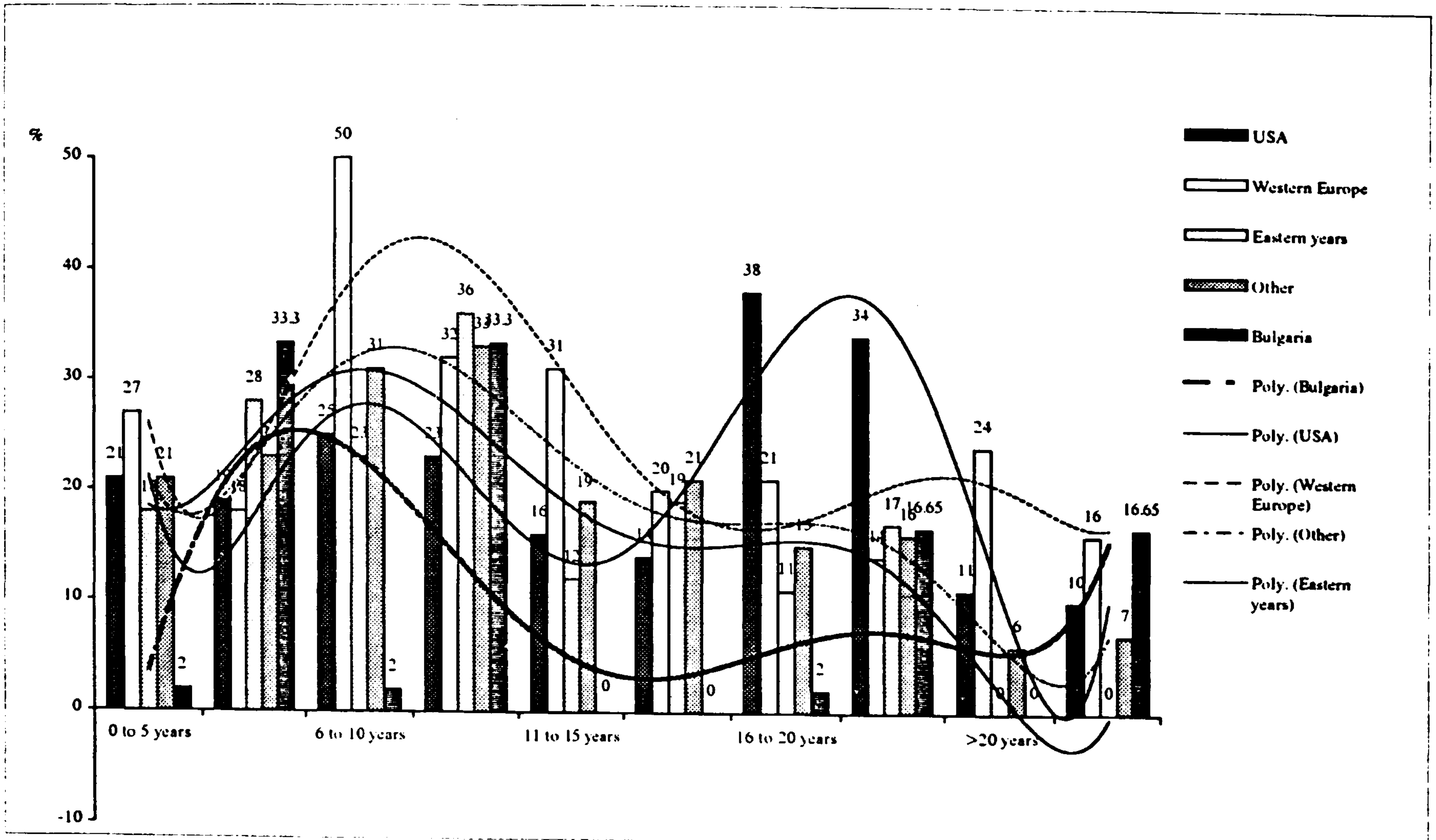


FIG.3. Distribution of the operating comercial reactors in the world depending on thier age

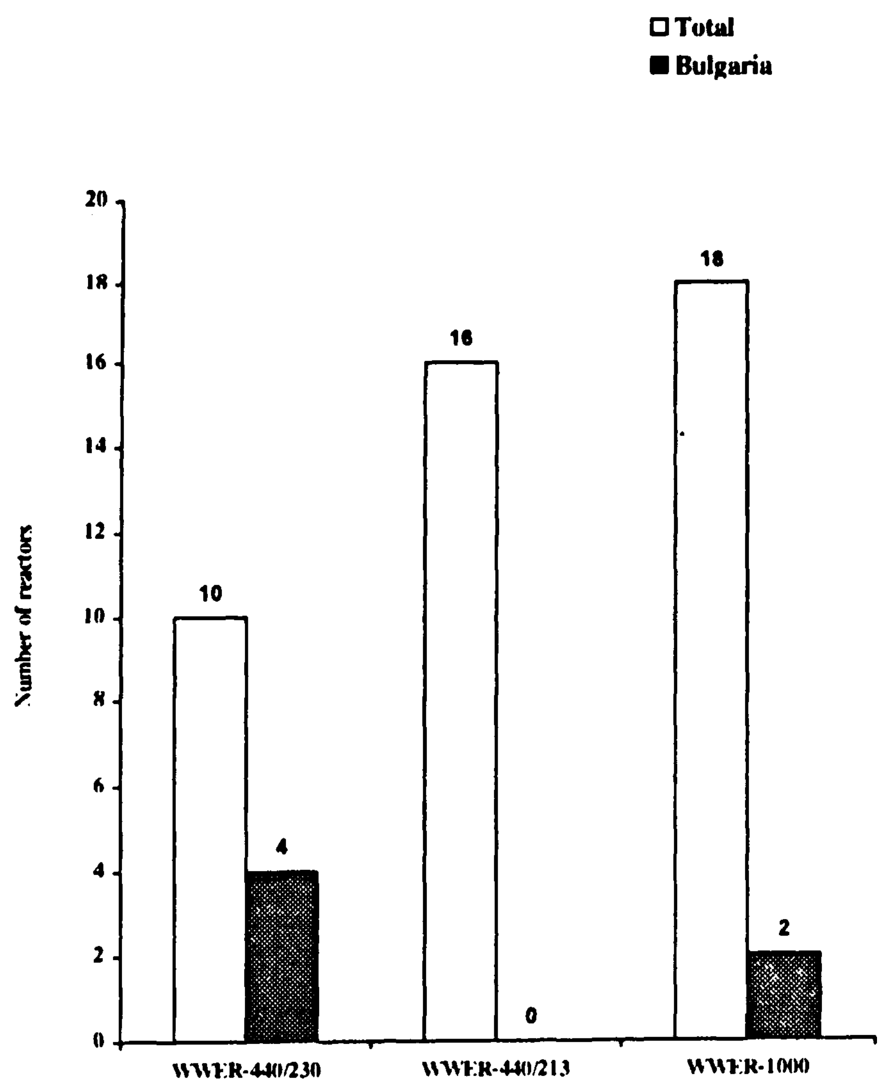
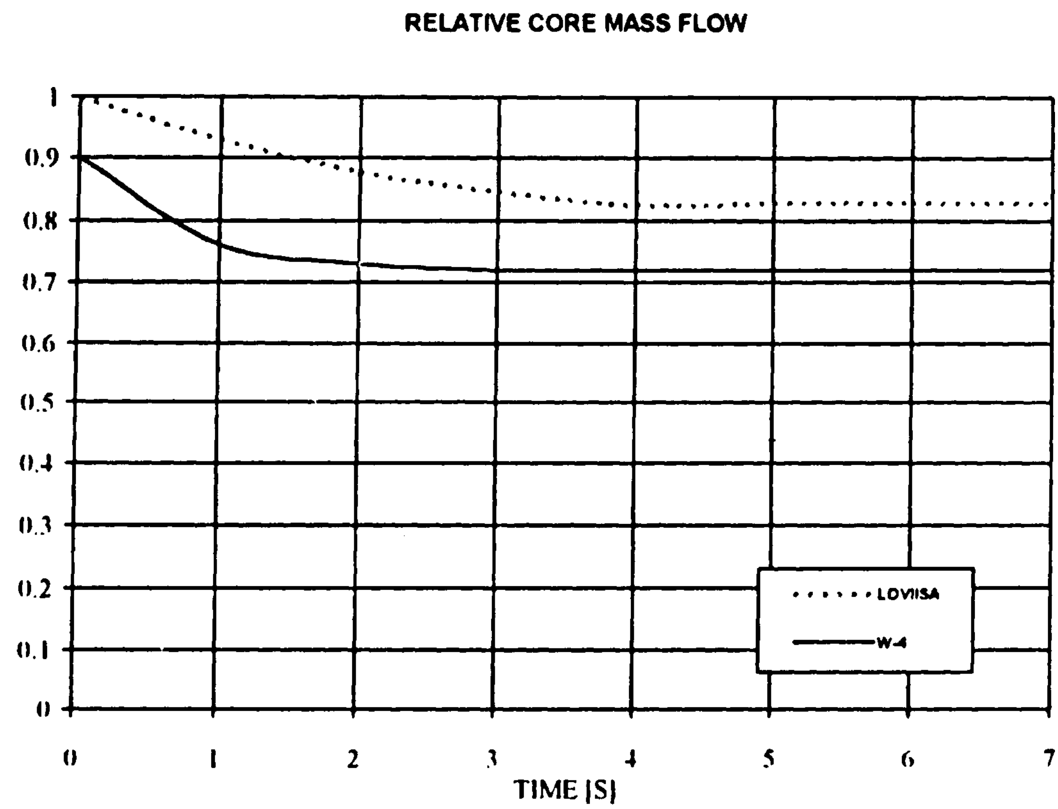


FIG.4. WWER-type of reactors in operation



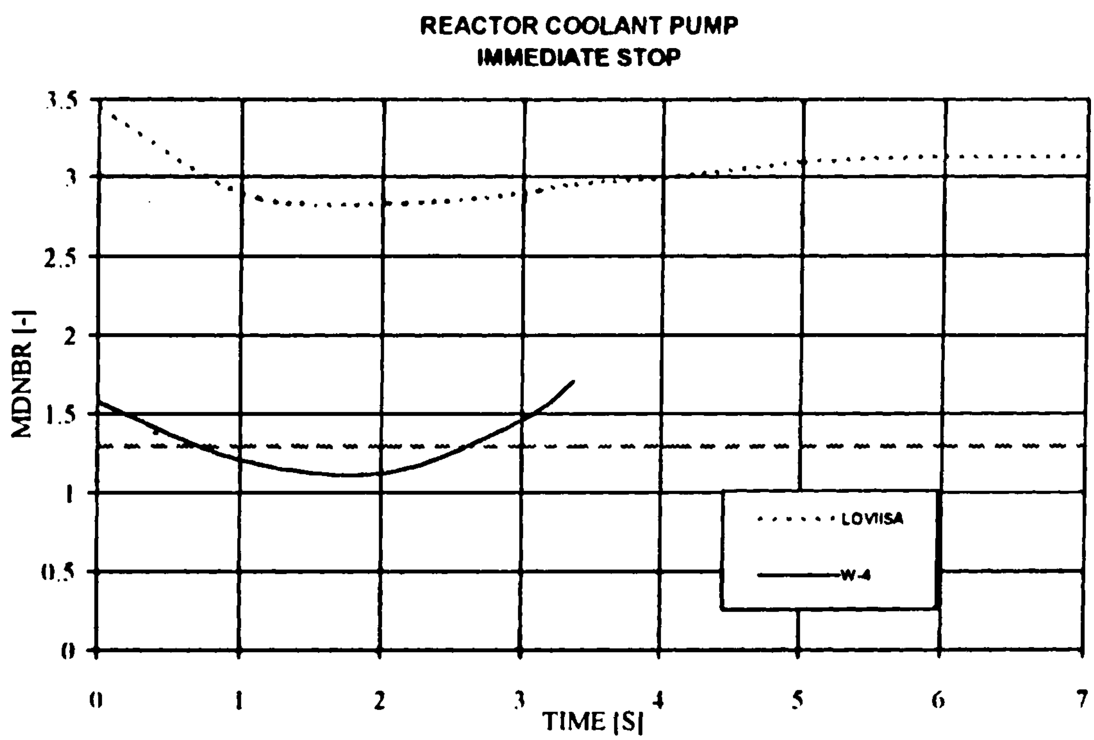
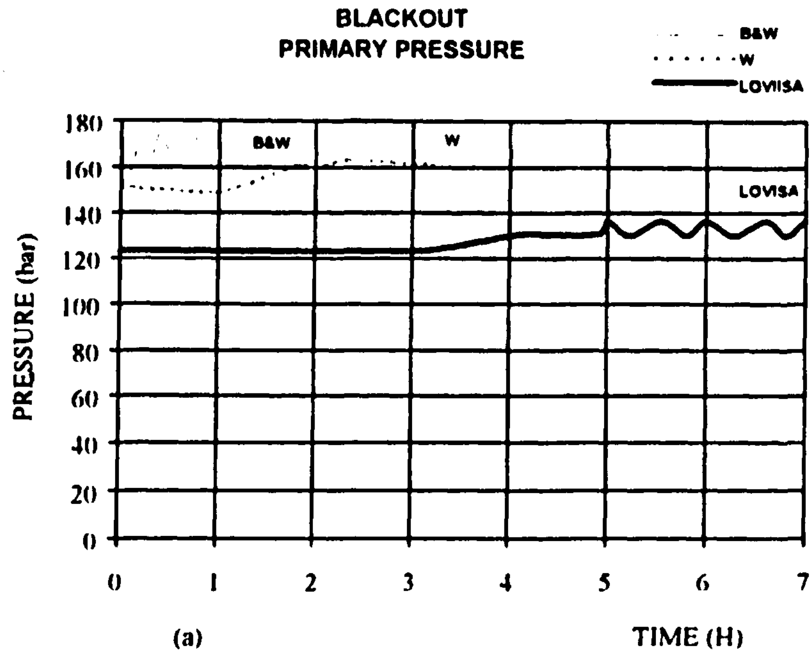


FIG. 5. Relative core mass flow rate and minimum DNDR in case of a reactor coolant pump seizure



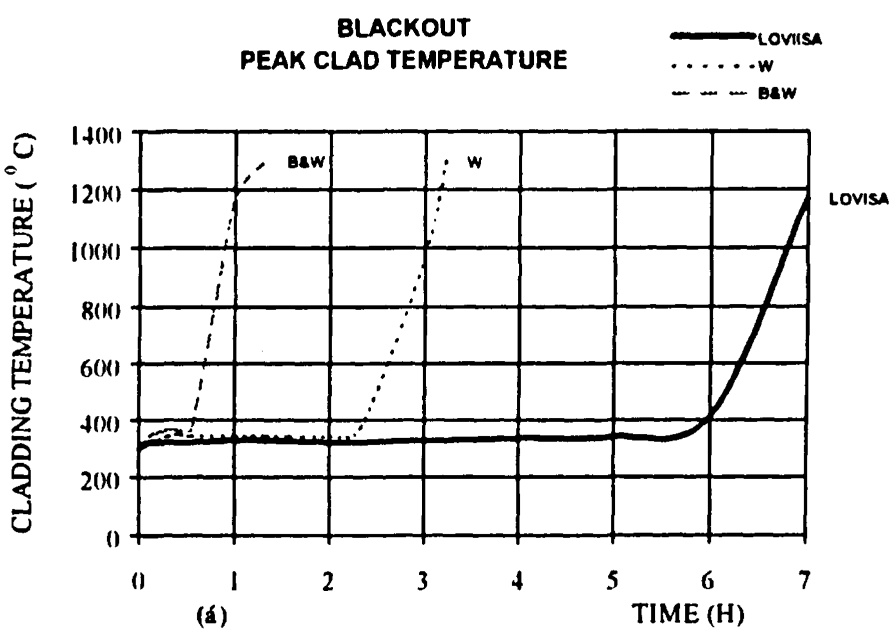
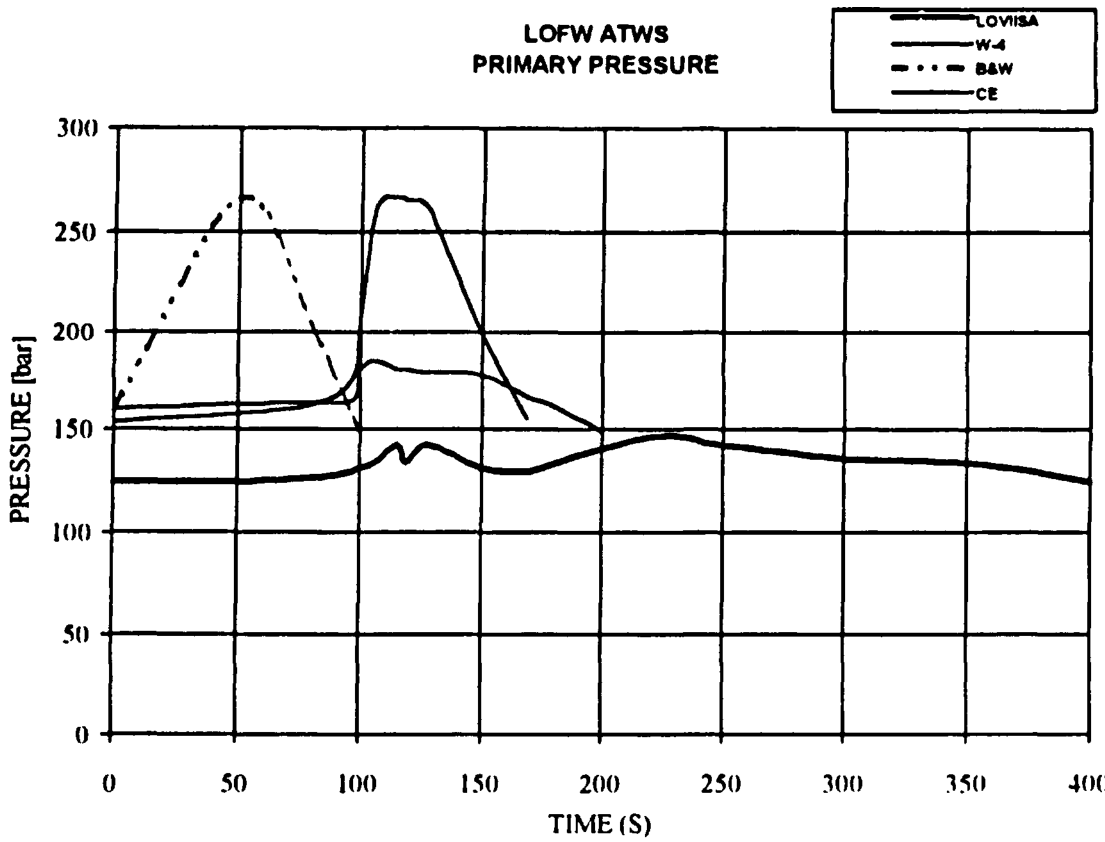


FIG.6 Primary pressure (a) and peak cladding temperature (б) after station blackout



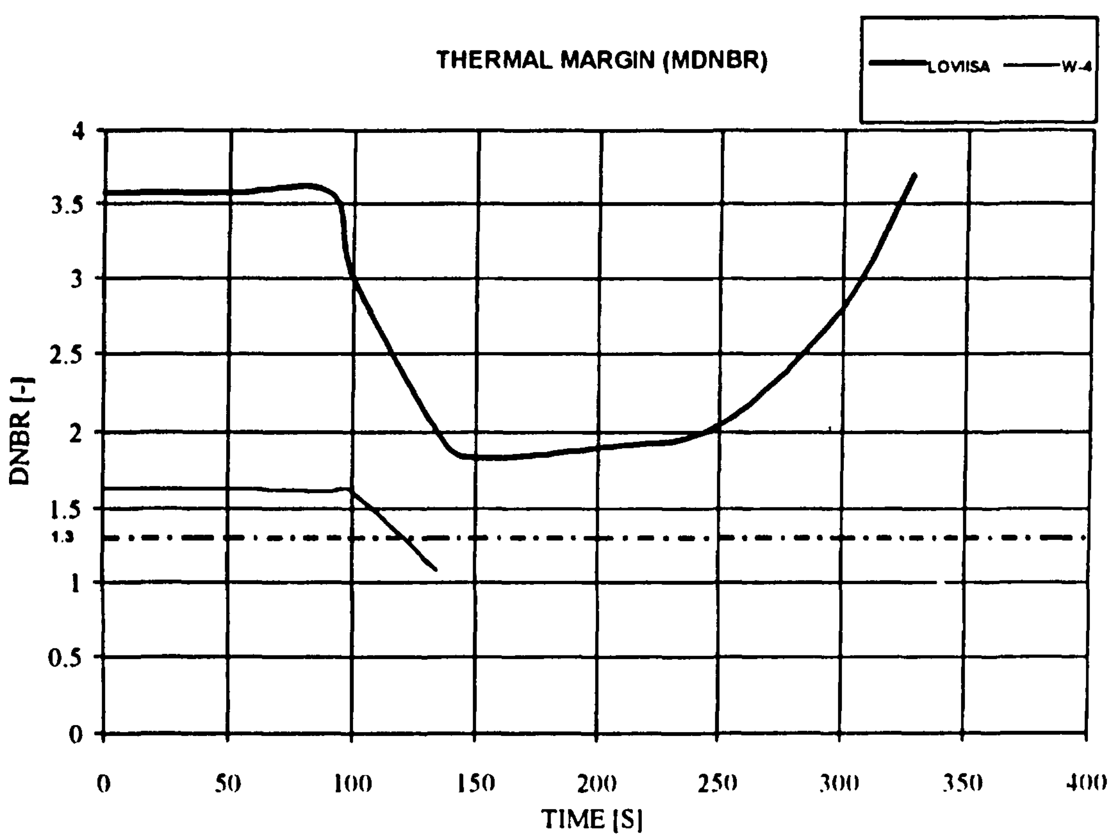


FIG.7. Primary pressure (above) and minimum DNBR (below) after loss of feed water without reactor scram

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PROBABILITY SAFETY ASSESSMENT OF UNITS 5 & 6 IN NPP KOZLODUY

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Risk Engineering Ltd.

INTRODUCTION

The Probability Safety Assessment (PSA) Level 1 of units 5 & 6 in NPP Kozloduy is a result of more then 2,5 years study done under a contract between NPP Kozloduy and Risk Engineering Ltd. It is practicaly the first investigation of this type for the russian type reactors VVER-1000 including internal accident initiators as well external initiators like earthquickes and fires. The study is based on a methodology reccomended by the International Atomic Energy Agency (IAEA) and different stages were checked by international review missions including final IAEA IPERS mission.

METHODOLOGY

The safety of nuclear power plants had always priority in comparison with other industries from the very beginnig of nuclear industry development. During the last years, however, significant opposition was formed against the use of nuclear power plants (NPP). This is probably connected not only with the risk of radiological release in case of nuclear accident, but also with social and psyhological factors. The growing number of NPP's leads to more stringent safety criteria. Also it should be taken in to account the need of high technological knowledge and culture of the processes in NPP, understanding of the effects of the radiological contamination of the environment. For many years these factors were underestimated in our country which transformed the safety problems in NPP in a political problem.

In the mean time the practice of the developed contries showed that the society can be convinced in the benefit of the NPP and it can accept the risk of the nuclear accident consequences if this will be done on the base of well established scientific methodologies and the results are understandable for the society. The last can be done on the base of comparison with the risk from other technological and/or nature hazards. Recently such estimations and analyses are done successfully with the methods of PSA which are well developed and established and are already a standart tool for NPP safety evaluation inmany contries.

The main benefit of PSA is to provide deep insights into plant design, performance and environmental impacts, including the identification of dominant risk contributors and the comparison of options for reducing risk. PSA provides a consistent and integrated model of nuclear power plant safety. Consequently, PSA offers a consistent and integrated framework for safety related decision making. Changes or alternatives in different design and engineering areas in a nuclear power plant can be compared on a common basis, namely the quantitative estimate of risk provided by PSAs. Furthermore, PSA is a conceptual and mathematical tool for deriving numerical estimates of risk for nuclear plants and industrial installations in general.

PSA is different from traditional deterministic safety analysis in that it aims at completeness in identifying accident sequences that can follow a broad range of initiating events and it requires the systematic and realistic determination of accident frequencies and consequences. A major advantage of PSA is that it allows for the quantification of uncertainties in safety assessments together with the quantification of expert opinion and/or judgement. Finally, PSA has been shown to provide important safety insights in addition to those provided by deterministic analysis.

The IAEA PSA Guidelines [1] were accepted as a main methodological basis internationally recognised.

In international practice three levels of PSA have evolved:

Level 1: The assessment of plant failures leading to the determination of core damage frequency.

Level 2: The assessment of containment response leading, together with Level 1 results, to the determination of containment release frequencies.

Level 3: The assessment of off-site consequences leading, together with the results of Level 2 analysis, to estimates of public risks.

Until now there was not performed full scope PSA for any unit in NPP Kozloduy as well as for any WWER type reactor in the world. This led to a loss of important technical information. After the project initialisation in 1992 on units 5&6 there was not any system for gathering and management of information for PSA purposes. This led to big difficulties because in parallel with the PSA study many support activities had to be maintained such as technical plant description several walk down procedures for gathering the information missing from plant documentation etc. The study was performed on the base of the recommended by IAEA documents [1, 2, 3] as well as on the base of the developed in the Risk Engineering Ltd Qualty assurance procedures.

SCOPE AND TASKS OF THE STUDY

The study covers PSA Level 1 according to IAEA PSA Guidelines [1], and additionally were included seismic and fire initiators. The scope was defined first of all by the contract terms of refernece and by the PSA procedures. The scope includes all significant accident initiators list of which was discussed during the first two phases of the study.

The experience gained during the fulfilment of the first phase as well as the recommendations from the first IAEA missions led to redistribution of the task in different phases and to their adjustment. This was caused by the following factors: the need of the additional thermohydraulic calculations and the need of additional seismological investigations.

As a result the final distribution of the phases was as follow:

1. Information gathering for PSA models and calculations
2. Accident sequences from internal initiators and fire hazard analysis
   - 2.1. Modelling of accident sequences from internal initiators and thermohydraulic analyses.
   - 2.2. Analysis of accident sequences from fire hazards.
3. Accident sequences from earthquake initiators
   - 3.1. Data preparation (seismic hazard assessment and structure response)
   - 3.2. Plant systems and accident sequence analyses
4. Final report preparation

So defined phases cover practically all procedural steps, described in the next section with methodology and organization and additionaly anlyses of the external initiators were performed.

The definition of the safety systems included in the PSA study was based on the assumtion (according to the terms of reference) that only the reactor core is analysed as a source of the radioactive materials during the work on 100% nominal power.

GENERAL PROCEDURAL STEPS OF A LEVEL-1 PSA

The six major procedural steps are:

- Management and organisation

- Identification of radioactivity sources and accident initiators
- Accident sequence modelling
- Data acquisition, assessment and parameter estimation
- Accident sequence quantification
- Documentation of analysis, display and interpretation of results

The general flow of work/information among these steps is shown in Fig. 1 and briefly described here. It is important to recognise that this flow is not always linear and that there are many iterative loops among the various steps.

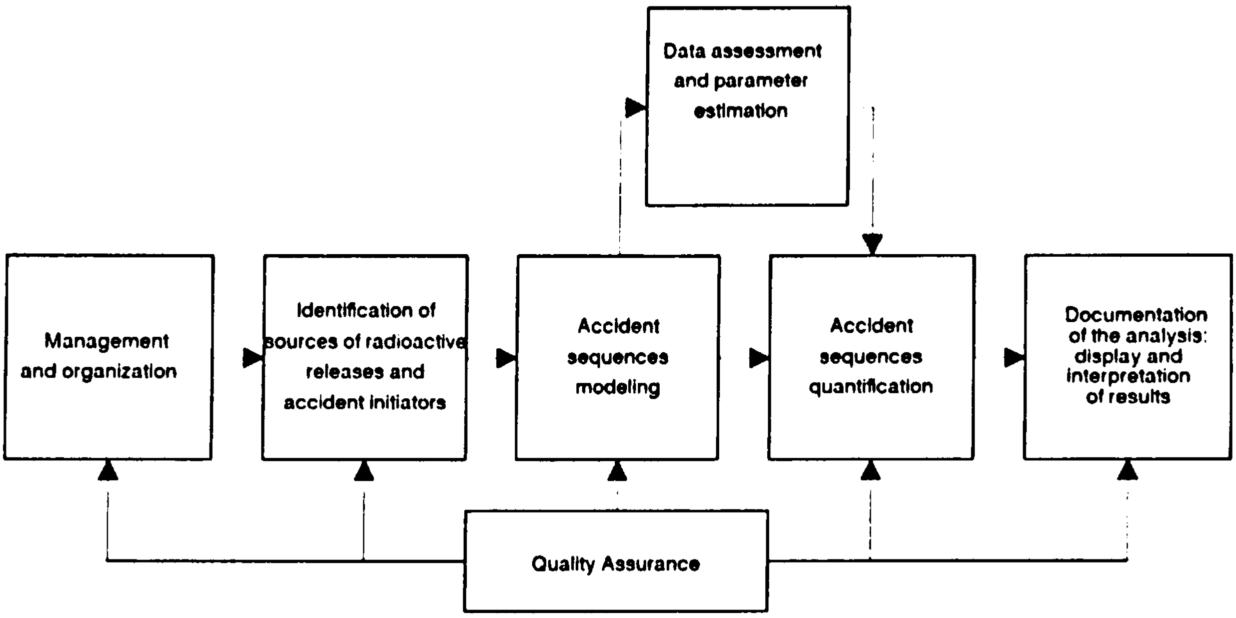

FIG. 1. Major procedural steps of Level 1 PSA

MANAGEMENT AND ORGANISATION

This step includes the actions and activities necessary for the organisation and management of the study. It includes the definition of the objectives, the scope, the project management scheme of the PSA, the selection of the methods and establishment of procedures, the selection of personnel and the organisation of the team that will perform the PSA, the training of the team, preparation of a PSA project schedule, the estimation and securing of the funds necessary, and the establishment of quality assurance and peer review procedures.

The project management and organisation as well as the key personnel were provided by Risk Engineering Ltd, but during the different project phases were involved several external institutions and many specialists. The main work connected with the information gathering was done by NPP specialist. Thermohydraulic analyses of the missing transients were provided by the group of E. Balabanov from "Energoproekt" - Sofia. In the beginning of the project there was not Safety Analyses Report (SAR) for units 5&6. Such document appeared at the end of 1993 and was used as an information source. The seismic studies for preparing data for seismic PSA were subcontracted to the group of M. Kostov from the Central Laboratory for Seismic Mechanics and Earthquake Engineering.

In table 1. are shown the main resources (full time and short time) for the PSA activities for the whale time of the project. The resources for information gathering are not included in this table.

Table 1. PSA Team Key Personnel

	Personnel	
	Full time	Short time
Project management	2	-
System analysts	6	2
PSA quantification specialists	3	2
Human performance analysts	1	1
Data analysts	1	1
Thermohydraulic analysts	1	2
Quality assurance control	1	-
Seismic hazards analysts	2	3
Fire hazards analysts	2	2

Communications for discussions and information gathering were held weekly through visits of REL specialists to NPP "Kozloduy". The existing telecommunications between Sofia and Kozloduy are still not effective due to the poor quality of the signal.

During the first phase of the project were adjusted and approved the REL QA procedures and methodologies for different project phases.

IDENTIFICATION OF RADIOACTIVE SOURCES AND ACCIDENT INITIATORS

The potential sources of radioactive releases to the environment are identified, the potential states of the plant to be analysed are determined, and the safety functions incorporated in the plant are identified. The accident initiators that can challenge these functions as well as the systems that serve them are identified. The relationships between initiating events, safety functions and systems are established and categorised. During this step, the analysis team becomes familiar with the plant to be analysed and the methods to be used and collects much of the required information on which to base subsequent analysis.

ACCIDENT SEQUENCE MODELLING

The third procedural step deals with the construction of a model that simulates the initiation of an accident and the response of the plant. This model consists mainly of a combinations of events that comprises initiating events. both internal and external, system failures and human errors that will lead to the undesirable consequences. These combinations of events are called accident sequences and the objective of this step is to define them. Models for the detailed analysis of system failures and of human errors are developed. A qualitative analysis for inclusion in the models of possible dependencies is also performed in this step.

DATA ACQUISITION, ASSESSMENT AND PARAMETER ESTIMATION

This procedural step acquires and/or generates all information necessary for quantification of the model that was constructed in the third step. In particular, the fundamental elements of the plant model and the parameters that need to be estimated are identified. The data necessary to produce these estimates and the associated uncertainties are collected and treated appropriately. The parameters that are estimated can be divided into three major categories: frequencies of initiating events, component unavailabilities and human error probabilities. Parameters necessary for the modelling of potential dependencies among various events (initiating events, hardware failures or human errors) are also estimated.

ACCIDENT SEQUENCE QUANTIFICATION

In this step, the model (constructed in the third step) is quantified using the results of the fourth step. The result of this step is the assessment of the frequency of accident sequences. Normally this is accompanied by an assessment of the associated uncertainties. Sensitivity studies are made for the important assumptions and the relative importances of the various contributors to the calculated results is indicated.

DOCUMENTATION OF ANALYSIS AND DISPLAY AND INTERPRETATION OF RESULTS

The results of the analysis are thoroughly documented in each step. In this step the results are displayed in the way that best meets the needs of the end users. This includes the interpretation of the results, in line with the objectives of the PSA.

PHASES OF THE STUDY

The phases of the study are defined by the contract terms of reference and PSA procedures.

1. Information gathering for PSA models and calculations

This was one of the most difficult phases as it was mentioned in the section "METHODOLOGY".

The safety functions and safety criteria were adjusted and the following base definitions were established based on [4]:

- Design limits
- Design base accident
- Design base earthquick (6-th degree on 12-degree scale)
- Maximal design base earthquick
- Maximal design base accident
- Safety of NPP
- Operational safety limits
- Conditions for safety operations
- Controled limits
- Transients with violation of normal operation
- Main reactor states
- Safety functions and safety criteria

A list of safety systems and safety important systems was prepared.

Procedures for gathering information for PSA Level 1 were developed, approved and put in operation.

2. Accident sequences from internal initiators and fire hazard analysis

2.1. Modelling of accident sequences from internal initiators and thermohydraulic analyses.

During this phase were developed event trees (ET) covered all intiators and fault trees (FT) covered all evaluated systems. For some scenaria were performed thermohydraulic calculations for adjustment of the success criteria. The analysis was done by the group of E. Balabanov from ENERGOPROEKT-Sofia with RELAP 5. 12 ET were developed which covered all groups of initiators. More then 700 FT were developed for all front line and support systems.

2.2. Analysis of accident sequences from fire hazards.

This was the longest phase where the main efforts were put toward gathering information for fire hazard PSA. One IAEA workshop and one IAEA review mission were held on this problem and the recommendations were taken in to account in the final report.

3. Accident sequences from earthquake initiators

This phase was splitted in two parts:

3.1. Data preparation (seismic hazard assessment and structure response)

An additional project was initiated and fulfilled by the group of M. Kostov. The results from this project provided the necessary input for seismic PSA.

3.2. Plant systems and accident sequence analyses

The ET and FT models for internal initiators were modified with respect of the seismic initiator and quantification was performed.

4. Final report preparation

The final PSA report has Summary Report, Main Report and Appendixes to the Main Report and the preparation goes in reverse order: first are prepared Appendixes, then - the Main Report and finaly - the Summary Report.

The Summary Report provides general view on the motivation, objectives, assumptions, results and conclusions of the study and can be used as a reference by a wide range of specialist on NPP safety as well as for the purposes of the final pier review.

The Summary Report has section with the structure and organisation of the PSA report documentation where detailed description of the sections in the Main Report and Appendixes is given. Here is also given the interrelation between different parts of the PSA documents.

As it was mentioned previously, the Summary Report is prepared last, inspite that it is the first part of the PSA documetation.

The main report should describe in depth all tasks of the PSA including the tasks related to organization and management. In addition, the main report should provide the necessary links between different parts of the report and the appendices to help the reader to locate any additional information on specific issues of interest.

The appendices are for material whose bulk and level of detail are such that its inclusion in the main report is unwarranted. As was pointed out earlier in this section, all information necessary to document the analysis is included in the main report. The appendices contain major parts of the descriptions of the plant and systems, important assumptions, detailed models, the system fault tree models, the complete database and a substantial part of the results (both final and intermediate). Not all the information in the appendices is in printed form. Some of it is appropriately stored in magnetic records (computer tapes or diskettes, word processor files, and so on).

The tasks connected with the information gathering as a rule were performed in one phase. But several tasks, connected with the modeling of FT and ET were done during several phases with itterative performance.

Practically all phases were passed through independent expert review.

RESULTS OF THE ANALYSIS

As a result from the itterations done during the anlyses of the initiators and takong in to account the recommendations of the IAEA mission conntected with this process, the grouping of the initiators was performed and the initiators groups were defined.

14 initiators were investigated covering practically all groups:

(1) Large LOCA: Leakage through equivalent diameter Dy > 300 mm;

(2) Intermediate LOCA: Leakage through equivalent diameter Dy = 125 - 300 mm;

(3) Small LOCA: Leakage through equivalent diameter Dy = 30 - 125mm;

(4) Very small LOCA: Leakage through equivalent diameter Dy < 30;

(5) Control rod ejection and LOCA.

(6) Small LOCA outside containment.

(7) SG collector header cover rupture.
(8) Single SG tube rupture.
(9) Spurious Reactor trip.
(10) Reactor trip without secondary circuit.
(11) Reactor trip with containment isolated.
(12) Loss of off-site power.
(13) Loss of service water.
(14) SGs overcooling.

The connections between the necessary functions and the systems which can maintain these functions (front line systems) are given in Table 2.

Table 2. Systems maintaining safety functions in NPP Kozloduy 3

	Safety functions	Frontline systems
1	Reactor trip	Control rods system (CY3); Make-up system (TK) and system for adding of reagents (TB10); High pressure injection system (TQn3,4, n=1,2,3); and emergency gas removal system (YR) or pressurizer relief valves system (YP21,22,23).
2	Primary circuit safety functions against overpressure	Pressurizer relief valves system (YP21,22,23); Emergency injection system (YP11,12,13); Emergency gas removal system (YR).
3	Material balance of the primary circuit	Make-up system (TK); High pressure injection system (TQn3,4, n=1,2,3); Emergency core flooding system (YT11,12,13,14); System for emergency and planned cooling (TQn2, n=1,2,3).
4	Residual heat removal in hot conditions	Turbine condensers (SD11,12,13); Steam dump facility to the atmosphere (RC11,12); Auxiliary feedwater pumps (RL51,52); Steam dump facility to the condenser (TX50,60,70,80S05); SG emergency feed water system (TX10,.20,30); During LOCA: see item 3.
5	Providing deep subcriticality during cooling	Make-up system (TK) and system for adding of reagents (TB10); High pressure injection system (TQn3,4, n=1,2,3); and emergency gas removal system (YR) or pressurizer relief valve system (YP21,22,23).
6	Cooling	See item 4.
7	Keeping the containment integrity	Containment spray system (TQn1, n=1,2,3); Containment isolating valves system; ventilation system TL21.
8	Long term cooling	Low pressure injection system (TQn2, n=1,2,3).

Further for the purposes of the unit behaviour and system unavailability modeling were defined dependencies between front-line and support systems as well as between support systems.

Evont trees were builded for the listed initiators. In order to simplify and unify the process of linking the event trees and fault trees in "Risk Engineering" Ltd. is developed a "Guide for Construction and Quantification of FT". All models of the covered by the study systems are developed using the recomendations of the "Guide ...".

More then 700 FT were developed including all front line systems, all support systems and some important for safety systems. Every system was investigated in terms of its functions, design characteristics and operating conditions.

Common assumptions during the modeling of the system failures are connected with the definition of the boundary conditions and success criteria, unavailabilities due to repair and tests, common cause failures, human errors and level of detailes.

It should be clearly mentioned that main attention was paied to the adequate modeling and searching for week points on the base of the comparative analysis rather then to the absolute probability characteristics. The reason for that are the specific issues of this study mentioned in the introduction as well as the fact that just recently was initiated a project for development of the system for development of a plant specific reliability data base.

The main results from the analysis are showing the following distribution between different initiators in the total core damage frequebcy:

- internal initiators - 85%;
- seismic initiators - 5%;
- fire initiators - 10%.

The main contributor from the internal initiators is the Loss of Ofsite Power with frequency 1.1×10^{-4}/y.

The same order of magnitude has "Small Small LOCA", but the reason for that is the frequency of the initiating event equal to 1 in this case due to the lack of reliable data. Some evaluations exist in which this frequency is 10^{-2} -10^{-3}, and in this case the core damage frequency is in the range of 10^{-6} - 10^{-7}.

On the level of basic events the main contributor are.the human errors and common cause failures.

PRELIMINARY CONCLUSIONS AND RECOMMENDATIONS

The main preliminary conclusion from the study is that the safety functions (SF) of WWER-1000 are successfuly covered by the safety systems (SS) and by important for safety systems.

Some preliminary recommendations can be done for the support systems of SS through which could be violated significantly the success criteria of the SF. As an example can be used the possibility of a wrong signal from a pressure gauge in the primary loop to isolate the system YT 11-14 B01.

It is necessary in the future to perform an additional study of the common cause failures for more precise quantification of their contribution. This and other related studies can be done in the frame of the new project for a "Living PSA for Units 5&6" which will use as a base the models and the results from "PSA Level 1" project.

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АЭС „КОЗЛОДУЙ“: 20 - ЛЕТНИЙ ОПЫТ ЭКСПЛУАТАЦИИ

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АЭС - Козлодуй

Основные этапы сооружения и ввода в эксплуатацию

Начало ядерной энергетике в Болгарии было положено подписанным в 1966 г. соглашением между правительствами Республики Болгария и бывшим СССР о строительстве и введении в эксплуатацию первой атомной станции в Болгарии. Отсюда начинается историят АЭС-Козлодуй.

Строительство АЭС-Козлодуй было начато земляными работами 06.04.1970 года на I энергоблоке и закончено 30.12.1993 года принятием последнего VI энергоблока в промышленную эксплуатацию. Строительство велось по русским (советским) проектам и технологии.

Строительство и ввод в эксплуатацию АЭС-Козлодуй включали три этапа, которые характерны постоянным повышением степени безопасности и экономической эффективности энергоблоков в соответствии с международными тенденциями в области ядерной энергетики. Первый и второй этапы охватывают строительство и освоение мощности I÷IV блоков, оснащенных реакторами ВВЭР-440/В-230/. Третий этап охватывает сооружение и освоение мощности V ÷ VI блоков, с реакторами ВВЭР-1000/В-320/.

Первый этап включал пуск и освоение мощности I и II энергоблоков. Он включал проведение послемонтажных испытаний оборудования и систем, вывод реакторов в критическое состояние и освоение проектной мощности с подтверждением безопасности и надежной эксплуатации энергоблоков на промежуточных уровнях. Вывод реактора I блока в критическое состояние произведен 30.06.1974 года, включение первого турбогенератора в энергосистему - 14.07.1994 года. Первый этап ввода в эксплуатацию закончен 07.11.1975 года по окончании 72 часовых испытаний II энергоблока на проектной мощности. Проектная мощность I блока освоена за 100 дней, а II блока - за 39 дней.

Второй этап начался в октябре 1973 года с началом строительства III и IV блоков и закончен 25.06.1982 года с принятием в промышленную эксплуатацию IV энергоблока. Физический пуск реактора III блока был осуществлен 04.12.1980 года, а энергетический - 16.12.1980 года. III блок достиг своей проектной мощности 27.01.1981 г., а IV блок - 27.06.1982 г. Освоение проектной мощности на III блоке достигнуто за 43 суток, а на IV - за 32.

Третий этап начинался 09.07.1985 года строительством V энергоблока и закончен 30.12.1993 г. принятием в промышленную эксплуатацию последнего VI блока АЭС-Козлодуй. V блок выведен в критическое состояние 05.11.1987 г., а VI блок - 29.05.1991 года. Эти блоки имеют современные системы безопасности, а их технологические процессы управляются автоматизированными компьютерными системами.

С введением в эксплуатацию VI блока завершено строительство АЭС-Козлодуй. При установленной мощности 3760 MW станция стала самым большим электропроизводственным предприятием в Республике Болгария, вырабатывая более, чем 40% всей электроэнергии в стране.

Технико-экономические показатели

Эксплуатацията АЭС-Козлодуй в период 1974 - 1994г. характеризуется надежной и устойчивой работой систем и оборудования. Проектные показатели станции полностью подтверждены проведенными гарантийными испытаниями отдельных блоков. Обобщенные технико-экономические показатели работы отдельных блоков за 20-ти летний период работы АЭС-Козлодуй близки к проектным, причем различия, особенно за последние несколько лет, определяются режимом работы национальной энергосистемы и необходимостью проведения продолжительных реконструкций и модернизации для повышения безопасности и надежности станции.

Отклонение некоторых показателей от проектных объясняется различием между реальными эксплуатационными условиями и принятыми в проекте. Различными являются, например:

- реальная и проектная температура охлаждающей воды конденсаторов;
- обеспечение теплом площадки АЭС и города Козлодуй, как дополнение к основному проекту;
- постоянное загрязнение конденсаторов турбогенераторов 1 - 7 из-за неэффективной работы систем шариковой очистки.

Производство электроэнергии

В ежегодном производстве электроэнергии отмечается устойчивый рост за исключением периода 1991-1992 г., когда II и I блок решением Правительства были остановлены для реконструкции и модернизации. В целом, с 1974 г. до конца сентября 1994 г. АЭС-Козлодуй произвела 205 млрд. 969 млн. квт.ч электроенергии. Только за 9 месяцев 1994 г. было произведено 10 млрд. 383 млн. квт.ч. Доля производства АЭС в совокупном производстве Национальной Электрической Компании за последние годы также имеет устойчивый высокий рост . В 1992 г. она составляла 37.4% , в 1993 г.- 41.9 % а за 9 месяцев 1994 г. - 44.4%. Таким образом АЭС-Козлодуй стала основным и определяющим энергопроизводителем в Республике Болгария.

Если рассматривать производство электроэнергии по годам за весь 20-летний период, можно проследить развитие и стабилизацию эксплуатации блоков АЭС Козлодуй. Даже без учета выработки электроэнергии в год пуска и второй год эксплуатации видно, что выработка электроэнергии на IV-ом блоке значительно выше, чем на I-ом блоке. Это объясняется выросшим опытом персонала и учетом многих предыдущих ошибок. Это же подтверждается и тем фактом, что выработка электроэнергии на различных блоках (без V и VI) достигает приблизительно одинакового уровня за различное время: I блок - на пятом году эксплуатации; II блок - на третьем году; III и IV - на втором году. Тенденция увеличения выработки электроэнергии отдельными блоками сохраняется долгое время, приблизительно 8-10 лет, после чего начинается спад.

Самая большая выработка электроэнергии на I и II блоках была в 1985 году, когда все четыре блока с реакторами ВВЭР-440 произвели 13 млрд. 131 млн квтч. Последнее самое высокое производство электроэнергии на III и IV блоках было в 1988 году. Снижение производства связано с увеличением продолжительности ремонтов по восстановлению технического состояния оборудования, проведением реконструкции и модернизации, а в последнее время - и из-за диспетчерских ограничений.

Что касается V и VI блоков, выработка электроэнергии по годам отличается известной неравномерностью. Если за первые 3 года отмечается снижение выработки электроэнергии на V блоке, то на VI блоке она непрерывно растет. Это различие объясняется фактом, что пятый блок является первым из серии ВВЭР-1000, и многие конструкторские и проектные недостатки были устранены на шестом блоке до его пуска. Необходимо также отметить, что в последние три года - 1992-1994 энергосистема Республики Болгария была не в состоянии принять полную нагрузку этих блоков, что со своей стороны вызывало вынужденный останов одного из них или работу на пониженной мощности и, как следствие, низкую годовую выработку электроэнергии. Такое состояние вряд ли изменится в ближайшие несколько лет, и потому от этих блоков нельзя ожидать значительного роста электропроизводства.

Расходы электроэнергии на собственные нужды

Расходы электроэнергии на собственные нужды атомной станции в наибольшей степени зависят от диктуемого ей энергосистемой режима работы. Они близки к проектному показателю только тогда, когда блоки работают приблизительно на 100% своей номинальной мощности. При введении диспетчерских ограничений (работы на пониженной мощности) этот показатель значительно ухудшается.

За период 1974-1989 г., характеризующийся полной загрузкой мощностей, собственные нужды были близки к проектным независимо от того, что были включены дополнительные объекты-потребители, не предусмотренные в генеральном проекте, такие как общестанционный корпус (ОСК), химводоочистка (ХВО), береговая насосная станция (БНС), администрация.

Первый блок за весь период эксплуатации использовал на собственные нужды около 7,6% от общей выработки при проектной величине - 7,32%. С учетом всех вышеуказанных потребителей собственные нужды были ниже проектных значений. Исключение составляет год ввода в эксплуатацию и год пуска после реконструкции, когда собственные нужды возрастали до 15% . Второй блок был более экономичным даже и в пусковые годы, когда расход электроэнергии на собственные нужды достигал 12%. В другие годы его потребление было около 7,5%. В период нормальной эксплуатации III-го и IV-го блоков их собственные нужды составляли около 7,5%. В это число не входят значения собственных нужд (С.Н.) за годы пуска блоков и в период основного ремонта I-го и II-го блоков, т.к. их потребление распределялось между третьим и четвертым блоками. Следует еще раз отметить, что в собственные нужды I÷IV блоков включены также и собственные нужды береговой насосной станции (БНС), и потери при передаче электроэнергии в энергосистему, суммарная величина которых за последние два года составила около 1,5%. Для правильной оценки экономичности блоков необходимо, чтобы в последующие периоды эти собственные нужды учитывались как общестанционные для всей площадки.

Собственные нужды V и VI блоков в исключительно высокой степени зависят от уровня нагрузки и колеблются между 4,5% и 11%. Со времени энергопуска до настоящего момента эти блоки практически работали с полной нагрузкой незначительное время. По этой причине их суммарные собственные нужды (С.Н.) значительно превышают проектные показатели. Чтобы улучшить этот показатель необходимо оптимизировать режимы эксплуатации, изменив и оптимизировав перегрузку блоков.

За последние несколько лет расход электроэнергии на собственные нужды, выраженный в % от электропроизводства, значительно возрос из-за наложенных диспетчерских ограничений. Это особенно отразилось на блоках с реакторами ВВЭР-1000. В режиме работы 50% номинальной мощности собственные нужды этих блоков, выраженные в % , вырастают больше, чем в два раза. Подобное положение наблюдается и на блоках с реакторами ВВЭР-440, хотя и в меньшей степени.

Коэффициент полезного действия реакторных установок.

Коэффициент полезного действия реакторных установок, определенный в процессе гарантийных испытаний, соответствует проектному. Но реальные эксплуатационные значения существенно отличаются от гарантийных. Причинами этого являются и те, что указаны выше, и диктуемые энергосистемой режимы работы станции. При пониженных мощностях и одинаковых эксплуатационных условиях из-за низкого расхода пара через турбинные установки эффективность регенеративной системы резко ухудшается, а это влияет на коэффициент полезного действия (К.П.Д.) установок. В случае с реакторами ВВЭР-440 из-за наличия двух турбин, при работе на 50% мощности с одной турбиной снижение К.П.Д. не так существенно, в то время как на ВВЭР-1000 работа при 50% мощности ведет к снижению проектного К.П.Д. проблизительно на 4%.

Блоки с реакторами ВВЭР-440 за период эксплуатации показали достаточно высокие значения коэффициента полезного действия. До 1991 года I блок работал с К.П.Д. 30.1%. В среднем за весь период эксплуатации, исключая 1992 год, коэффициент равнялся 29,8%. II блок имеет средний К.П.Д. за срок своей эксплуатации 29,7%, а после 1992 года он равен 27.9% . Самый высокий коэффициент за срок эксплуатации у III блока - 30.9% , а затем у IV - 30,5%. На этих блоках рассчетная температура охлаждающей воды для конденсаторов турбин - 22 ^{o}C , а на I и II блоках расчетная температура - 12 ^{o}C.

На V и VI блоках коэффициент полезного действия в зависимости от режимов работы блоков колеблется от 29.55% до 33,08% . Самый низкий показатель для блоков ВВЭР-1000 был отмечен в 1990 году на V блоке, К.П.Д. которого за весь период эксплуатации изменялся в границах от 29,55% до 32.17%. VI блок значительно экономичнее и его коэффициент изменяется от 31.4% до 33.08%. Самые высокие значения были отмечены в 1993 году, когда блок работал преимущественно на мощности выше, чем 75% от номинальной. Анализ изменения коэффициентов по годам подтверждает тот факт, что при работе энергоблоков на- и около номинальной мощности достигаются более экономичные режимы и соответственно высокие значения К.П.Д. При снижении мощности на 50%, особенно на V и VI блоках, резко ухудшаются показатели экономичности - коэффициента полезного действия и собственных нужд. При существующем состоянии энергосистемы и её ближайших перспективах нельзя ожидать существенного увеличения нагрузки блоков, особенно в летнем режиме, а, следовательно, и улучшения показателей экономичности.

С другой стороны, энергоблоки с реакторами ВВЭР-440 летом могут достаточно экономично работать при 50% своей номинальной мощности, при этом будет обеспечена возможность полной нагрузки реакторов ВВЭР-1000. Это привело бы к существенному снижению уровня собственных нужд, снижению потерь из-за снижения К.П.Д. и, в конечном счете, к существенному увеличению произведенной электроэнергии (5-6%) при одних и тех же затратах на ее производство.

Средняя нагрузка блоков I÷IV с реакторами ВВЭР-440 за фактически отработанное время является приблизительно равной. На I, II и III блоках она составляет 375 Мвт, а на четвертом - 391 Мвт. I и II блок имели более низкую среднюю нагрузку за реально отработанное время в течение первых двух лет эксплуатации. Начиная с 1991 года III и IV блок продолжительное время работали на пониженных уровнях мощности из-за диспетчерских ограничений или работы на мощностном эффекте. В этот период I и II блоки были в основном на реконструкции.

Использование установленных мощностей

Плановые остановы I блока имеют среднюю продолжительность 707,5 часа, включая простой блока в 1991-92 году, при этом средняя продолжительность до этого простоя была 416,7 часа. Начиная с 1989 года наблюдается тенденция к увеличению времени плановых ремонтов за счет увеличения объема ремонтно-профилактических работ и контроля.

II блок останавливался планово в среднем на 324,6 часа до 1991 года и на 465,5 часов за весь период эксплуатации.

По III блоку это число - 502,6 часа. Объяснение увеличения длительности плановых остановов III блока в том, что блоки III и IV имеют дополнительные по сравнению с блоками I и II, системы и оборудование, связанные с безопасностью.

Средняя продолжительность планового простоя IV блока составляет 415,5 часов.

Плановый простой V блока за период его эксплуатации очень неравномерен, с цикличностью 2 ÷ 3 года. Объясняется это тем, что V блок является первым энергоблоком В-1000 нового типа, построенным за пределами бывшего СССР. За время эксплуатации выявлен ряд дефектов и неудачных проектных решений в технологических схемах и оборудовании. Это потребовало и все еще требует удлинения ремонтных кампаний и аварийных остановов блоков.

На VI блоке картина в корне отличается. С момента его пуска в 1991 году его показатели постоянно улучшаются: коэффициент использования установленной мощности (КИУМ) вырос с 29,3% (1992 год) до 69,4% (1994 год), при этом не следует забывать, что блок находился в процессе освоения проектной мощности до 30.12.93 год. До настоящего момента VI блок показывает очень хорошие результаты. В отличие от V блока, на VI блоке еще во време монтажа и наладки были устранены все дефекты и недостатки, проявившиеся при эксплуатации V энергоблока.

Коэффициент использования установленной мощности (КИУМ) за весь период промышленной эксплуатации для I блока в среднем составляет 73,2%, исключая годы пуска. Но в некоторые годы он был значительно выше среднего значения. Например, в 1985 году он достиг 84% . В начальный период эксплуатации значение этого коэффициента было от порядка 55% .

На II блоке среднее значение коэффициента использования установленной мощности составляет 74,65% . Самым высоким его значение было в 1978 году - 87,1% . В последние годы он колеблется около 69% .

На III блоке последние три года существенно повлияли на среднее значение коэффициента, составляющее 75,1% . До 1990 года он был около 79,2% . Максимальное значение было получено в 1985 году - 89,8% . В последние годы он снизился в среднем до 61,2%.

На IV блоке наблюдается подобная же тенденция. Средний коэффициент использования установленной мощности за период эксплуатации составляет 77,0% , при этом до 1990 года в среднем он был 81,9% . Самым высоким его значение было в 1984 году - 90,7% .

Повышенные требования к безопасности и изменившиеся условия со стороны энергосистемы за последние несколько лет привели к отказу от стремления к достижению высоких значений коэффициента использования установленной мощности (КИУМ). Политика АЭС по проведению реконструкции и модернизации в рамках расширенных основных ремонтов отражается на использовании мощностей по времени, но с другой стороны значительно повышает безопасность и готовность блоков, а также надежность оборудования и систем.

После 1989 года АЭС-Козлодуй была открыта для Миссий Международного агенства по атомной энергии (МАГАТЕ) - Вена. Первая проверка уровня безопасности, во время которой были констатированы недостаточные проектные гарантии ядреной безопасности I и II блоков с точки зрения современных требований, была проведена на I ÷ IV блоках в июне 1991 года. По этой причине решением Правительства I и II блоки были остановлены. После обобщения результатов этой и подобных проверок на других станциях с реакторами ВВЭР-440 был представлен доклад, в котором указывалось на недостатки, давались замечания и предложения по повышению безопасности, в частности, для блоков I и II. Они были систематизированы в два направления:

- техническое - состоящее из 8 групп, которое охватывает 61 проблему с конкретными замечаниями и рекомендациями,
- эксплуатационное - из 6 групп, охватывающее 36 проблем и 166 конкретных замечаний.

Для решения этих проблем АЭС-Козлодуй совместными усилиями болгарских специалистов и экспертов МАГАТЕ и ВАО АЭС разработала и уже выполняет долгосрочную и краткосрочную "Программу повышения эксплуатационной надежности и безопасности блоков с реакторами ВВЭР-440 (В-230) АЭС-Козлодуй, с конечным сроком исполнения - 1995 год.

Впоследствии "Программа.." была распространена и на III и IV блоки для поэтапной реализация в рамках плановых годовых ремонтов.

Выполнение краткосрочной программы на II блоке закончилось в декабре 1992 года, а на I блоке - в декабре 1993 года. Оба блока уже пущены в эксплуатацию при допустимом по современным требованиям уровне безопасности.

Все проекты и работы по реконструкции и модернизации I и II блоков выполнялись под наблюдением и были приняты Государственным надзорным органом - КИАЭМЦ, контролирующим эксплуатацию АЭС-Козлодуй с точки зрения ядерной безопасности и специально созданным ему в помощь международным Консорциумом надзорных органов по ядерной безопасности Франции, Бельгии, Германии и Великобритании, которые занимались актуализацией и оценкой каждой отдельной проблемы. Первая такая проверка АЭС - Козлодуй была проведена в апреле 1993 года, а вторая - в июле 1994 года.

С целью обеспечения безопасности была выполнена термообработка корпусов реакторов I, II и III блоков для снятия напряжений, вызванных облучением материала корпуса нейтронными потоками. Исследуются образци материалов, использованных для производства трубопроводов и сооружений, проработавшие длительное время (более 1000 тыс. часов) в условиях I контура. До настоящего момента полученные результаты положительны и позволяют предположить возможность эксплуатации блоков до выработки ими проектного ресурса.

Также в связи с новыми международными нормами безопасности была разработана программа комплексных мероприятий по приведению V и VI энергоблоков в соответствие с новыми нормативными требованиями к безопасности.

Программа содержит два раздела:

- долгосрочный, включающий исследования и проектирование, связанное с новой степенью сейсмичности площадки, и
- краткосрочный, включающий конкретные изменения в технологичных схемах и оборудовании.

В целом, за прошедшие 20 лет АЭС-Козлодуй показала безопасную, надежную и экономичную работу. Она выработала более 200 млрд. квт/часов электроэнергии и стала основным и решающим электропроизводителем, без которого немыслимо будущее энергетики Болгарии.

Непрерывное повышение безопасности и надежности - приоритетные цели последних лет - дают гарантию ее будущего развития и просперитета.

ПРОИЗВЕДЕНА ЕЛЕКТРОЕНЕРГИЯ [kWh]

Година	Блок 1	Блок 2	Блок 3	Блок 4	Блок 5	Блок 6	Общо
1974	928,241,650						928,241,650
1975	1,944,754,266	610,187,404					2,554,941,670
1976	2,250,140,381	2,738,606,058					4,988,746,439
1977	2,803,868,158	3,080,329,807					5,884,197,965
1978	2,551,983,704	3,358,630,450					5,910,614,154
1979	3,019,166,093	3,161,267,841					6,180,433,934
1980	3,079,747,042	3,071,695,932	13,235,346				6,164,678,320
1981	3,065,584,498	2,912,478,569	3,140,572,743				9,118,635,810
1982	2,901,560,590	3,018,483,733	2,874,666,772	1,950,924,188			10,745,635,283
1983	3,068,861,795	3,176,788,739	2,968,813,443	3,102,872,321			12,317,354,298
1984	2,978,551,808	2,868,073,796	3,383,332,733	3,505,412,577			12,735,370,914
1985	3,238,240,196	3,168,513,573	3,459,741,511	3,264,857,145			13,131,352,425
1986	2,740,294,770	2,999,644,418	2,910,557,446	3,420,055,547			12,070,552,181
1987	3,192,757,308	2,844,183,894	2,972,890,190	3,269,333,773	156,268,800		12,435,433,965
1988	3,072,831,409	2,141,393,601	3,376,575,947	3,134,648,653	4,304,592,000		16,030,041,610
1989	2,330,256,739	2,923,174,710	2,622,150,345	3,103,736,421	3,586,154,400		14,565,472,615
1990	2,747,665,400	2,642,972,863	2,827,110,075	2,783,894,247	3,663,187,200		14,664,829,785
1991	1,655,435,794	1,605,038,984	1,797,395,391	1,146,599,224	2,089,526,400	1,447,662,400	9,741,658,193
1992		4,699,012	2,603,424,614	2,563,207,781	3,788,798,400	2,592,281,760	11,5[illegible]2,411,567
1993	10,654,520	2,587,828,630	2,144,457,920	2,300,102,615	3,447,907,200	3,374,895,580	13.865,846,465
1994	2,782,569,708	1,880,105,808	1,183,864,251	1,207,563,249	3,061,258,800	5,185,784,136	
Общо	50,363,165,829	50,794,097,822	38,278,806,727	34,753,207,741	24,097,693,200	12,600,623,876	210,887,595,195

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ДВАДЦАТЬ ЛЕТ АТОМНОЙ ЭНЕРГЕТИКЕ РЕСПУБЛИКИ БОЛГАРИИ

(экономические и политические аспекты)

Никита Шервашидзе - Председатель Комитета энергетики

Двадцатилетний юбилей - это не просто юбилей, но и повод для подведения итогов, возможность обернуться с гордостью назад, подумать о будущем атомной энергетики.

Я не буду затрагивать никакие технические аспекты. Давайте попробуем найти место АЭС в контексте экономического и политического развития Болгарии.

Политический аспект проблемы, разумеется, очень деликатен, но за последние 3 - 4 года много было потрачено нервов и поломано судеб в попытке доказать, что энергетику ни в коем случае нельзя связывать с политической конъюнктурой (или развитием, если это более приемлимо), а тем более атомная энергетика не может быть связана с определенной политической и/или идеологической системой взглядов. Мне кажется, что эти спекуляции уже в прошлом, но не надо забывать и того, что были призывы лидеров политических партий закрыть АЭС „Козлодуй", воздвигнуть памятник коммунизму в Белене с корпусом реактора и другие невероятные истории, которые были реальностью. Старая русская поговорка гласит: "Кто старое помянет, тому глаз вон, а кто забудет, тому оба."

Обращаю ваше внимание на это не для того, чтобы бередить старые раны, а только для того, чтобы напомнить, что будущее атомной энергетики - это не только чисто экономическая или чисто техническая проблема. Все попытки свести проблемы только к этим двум ставят под вопрос будущее атомной энергетики, при этом не только в Болгарии.

В унисон с чисто политическими спекуляциями против атомной энергетики была и подлинная озабоченность людей, занятых в этой области, тем, что небрежная эксплуатация, феодальная близорукость и самодовольство достижениями АЭС могут быть (и были) причинами серьезного основательного беспокойства общества о последствиях самоуспокоения тех, от которых зависит безопасность.

Истинная катастрофа Чернобыля не ограничивается регионом действия. На карту было поставлено будущее АЭС, и еще один серьезный инцидент в АЭС был бы фатален для Болгарии.

Говорю об этом потому, что будущее атомной энергетики в Болгарии, а также и в мире - это нечто большее, чем себестоимость энергии, размещенность блоков или технические характеристики защитных систем, независимо от того, в какой степени они являются определяющими для наличия АЭС в энергетическом балансе Болгарии. Если мы забудем политику, снова окажемся в 1990 г.

Для атомной энергетики одинаково опасны как люди, склонные отрицать что-то, не попробовав его понять, так и люди, искренно убежденные, что без них атомная энергетика не может существовать, что они знают истину и что никто и ничему научить их не может. И если первую группу надо спокойно и без высокомерия убеждать, то наш долг противопоставить себя без компромиссов второй категории.

Много написано и пишут об экономике АЭС. Ведутся ожесточенные споры. Мы, как профессионалы, не должны сомневаться в том, что производим и будем производить самую дешевую электроэнергию. Почему, как, что будет, если сюда включить расходы на отработанное топливо, и захоронение не имеет значения для генерального вывода.

Что касается строительства (завершения) новой АЭС в Болгарии, то здесь надо семь раз отмерить, прежде чем предложить на суд широкой общественности и воле управляющих.

Здесь я только отмечу решающие для окончательного успеха будущего строительства АЭС моменты.

На первом месте стоит време, за которое мы сможем построить и пустить новый атомный блок. Деньги имеют цену во времени, и, если строительство продолжается более 5-6 лет, то АЭС становится нерентабельной.

Второй момент - это стандартизация и унификация проекта и выполнение. В некоторой степени это предопределяет, но не исчерпывает понятие "время".

Самое важное для нас - это форма финансирования проекта, т.е., кто и на каких условиях даст нам кредит и как мы убедим потенциального кредитора в том, что мы надежные партнеры, а правительство и парламент в том, что Болгарии без второй АЭС не обойтись. Здесь достаточно аргументов, но их нужно осторожно дозировать. Что нам легче всего экспортировать - помидоры, экологов или электроэнергию? Опыт эксплуатации АЭС дает нам основание претендировать на большее, по крайней мере, на Балканском полуострове. АЭС дает не только стабильность, национальную независимость и чувство собственного достоинства, но и экономическую основу для широкого поля действий. Более выгодными нам кажутся инвестиции в АЭС и экспорт электроэнергии, чем переброс электроэнергии на далекие расстояния. И, наконец, экология - это наука не только о природе, но и жизненном стандарте человека, т.е., о гармонии человек-природа. Мы не сможем поддерживать сносный для Европы стандарт, если будем импортировать электроэнергию, тем более после того, как мы доказали, что можем ее производить по самым современным технологиям, таким как атомная энергетика.

Разумеется, будущие государственные деятели Болгарии могут пойти путем Греции или Австрии, но им должно быть ясно, что это будет за счет и без того низкого стандарта нашего народа.

Знаю, что в том, что я говорю, нет ничего принципиально нового для энергетиков-специалистов, я просто хочу подсказать тон и аргументы, которые вне узкоспециализированных дискуссий, и с помощью которых пытаемся убедить более широкий круг заинтересованных лиц. Без этого атомная энергетика не может развиваться.

Энергетики часто нас обвиняют в повышенном чувстве собственного достоинства и, в общем, правы. Сложность техники, с которой работаем, процессы, которыми управляем, дисциплина и строгая организация, которой мы подчинены, дают нам основание для этого. Мы должны обязательно осознавать всю тяжесть ответственности, а это еще более важно для атомных энергетиков.

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ЯДЕРНАЯ БЕЗОПАСНОСТЬ, МЕЖДУНАРОДНОЕ СОТРУДНИЧЕСТВО И ПЕРСПЕКТИВЫ ДЛЯ БОЛГАРИИ

доц. к.х.н. Я. Янев, Председатель КИАЭМЦ

В прошедшие после 1955 г. 40 лет, когда в Женеве состоялась первая международная мирная конференция Организации Объединенных Наций " **АТОМ ДЛЯ МИРНЫХ ЦЕЛЕЙ**" использование ядерной энергии всегда имело международное измерение. Разделение мира на две сильные политические, экономические и военные системы оказало значительное влияние на развитие ядерных энергетических программ в различных частях света. Процесс, который мы сейчас наблюдаем, и в котором в той или иной степени участвуем последние несколько лет - это сильная интернационализация ядерных проблем и особенно проблем, связанных с повышением ядерной безопасности и культуры безопасной эксплуатации АЭС во всех странах Европы, Азии и Америки. В течение всех этих лет международное сотрудничество сыграло существенную роль в повышении безопасности и улучшении эксплуатации ядерных сооружений. Едва ли мы можем сомневаться или даже просто себе представить будущую работу на атомных станциях в мире без подобного сотрудничества. Более того, будущее развитие Болгарской ядерной энергетической программы в большой степени зависит не только от принятия ее нашим обществом, но и остальными странами Европы и мира.

Состояние ядерной энергетики в мире

За последние 50 лет, которые прошли от пуска первого ядерного реактора, которым было положено начало ядерной эпохи в развитии человеческой цивилизации, ядерная индустрия превратилась в ведущий научный и технологический потенциал развитых обществ. Уже здесь надо отметить, что ядерная энергетика - это характерная черта всех развитых государств, в том числе и Болгарии.

На сегодняшний день ядерная энергетика удовлетворяет 5% от суммарных энергетических нужд человечества, 7.5% всей электроэнергии мира производят атомные станции. К концу 1993г. в мире работали 425 ядерных блоков в 30 странах. В 1993 г. в мире было произведено более 2000 ТВч ядерной энергии общей стоимостью более 75 миллиардов долларов. Это равно электроэнергии, произведенной в мире за 1958 год.

Использование атомной энергии в Болгарии

Болгарская ядерная энергетическая программа начинается в конце 60-х, после подписания договора с Советским Союзом о доставке водо-водяных реакторов типа ВВЕР-440 модели 230. На территории Козлодуя за очень короткий срок, полностью соизмеримый с мировыми стандартами, построены первые два блока, а впоследствии, еще два, при этом 1 и 2 блоки - это типичные представители модели 230, а в 3 и 4 уже предусмотрены некоторые проектные решения модели 213. С экономической точки зрения можно уверенно сказать, что первая часть болгарской ядерной программы успешна. Если даже принять паритет рубля к доллару, то цена установленной мощности менее 600 долларов на установленный КВч, что является значительным достижением.

С другой стороны, развитие ядерной программы оставляет желать лучшего с точки зрения развития ядерного законодательства, системы лицензирования и обеспечения качества. В большой степени этим объясняются и некоторые неудачи, также как и состояние ряда все еще нерешенных проблем (таких как выбор места для постоянного национального хранилища радиоактивных отходов, нерешенные проблемы отработанного топлива), так и определенная неуверенность при выборе будущего АЭС.

Лишь в 1985 г., когда уже были подписаны договоры о поставке нового типа реактров - ВВЕР-1000, а пятый блок, реактор которого этого типа, находился в завершающей фазе строительства и монтажа оборудования, появляется **"Закон об использовании атомной энергии для мирных целей".** Этот закон создает возможность для регулирования и контроля атомной энергии, но к сожалению, снова не решает никаких кардинальных проблем, таких, как точное определение функций эсплуатирующей организации, контролирующего органа, а термин регулирование развития атомной энергетики даже не упоминается. Не смотря на некоторые существенные с современной точки зрения недостатки, "Закон об использовании атомной энергии для мирных целей" все-таки создает необходимый минимум условий для безопасного развития атомной энергетики. Основополагающие положения этого закона:

1. Государственная собственность, контроль и управление ядерными установками и ядерными материалами.
2. Единая система контроля и управления безопасным использованием атомной энергии.
3. Государственный контроль безопасного использования атомной энергии возложен на КИАЭМЦ во взаимодействии с другими специализированными государственными организациями.
4. Не смотря на то, что Закон довольно несовершен, в нем предвиден режим государственных компенсаций в случае ядерной аварии.

На базе ЗИАЭМЦ развиты и основные документы , обеспечивающие контроль безопасного использования атомной энергии (указы 2 и 8) относительно:

- сообщения об аномальных и аварийных событиях,
- основных правил безопасности,
- отчета, сохранения и транспортировки ядерных материалов,
- лицензирования при использовании атомной энергии,
- требований к обучению, квалификации и лицензированию персонала,
- сохранения, переработки и складирования РАО,
- физической защиты АЭС,
- противопожарной защиты АЭС.

Существенное значение имеют и "Основные нормы радиационной защиты" ОНРЗ-92, а также ряд других подзаконных актов.

В 1992, 1993 и 1994 годах специалистами страны была выполнена значительная работа по гармонизированию нашего ядерного законодательства. Разработанный и принятый Советом Министров "Закон об изменении и дополнении ЗИАЭМЦ" устраняет основные недостатки закона 1986 г.

Особое значение имеет принятие парламентом Венской конвенции и Совместного протокола о применении Венской и Парижской конвенций. Несмотря на то, что парламент не принял изменение ЗИАЭМЦ, посредством ратификации Венской конвенции мы значительно улучшили и свое внутреннее законодательство на основе приоритета международных договоров I силу конституции. В этом смысле, подписывание Конвенции о ядерной безопасности в сентябре 1994 г. является значительны шагом нашего ядерного законода-тельства.

Международное сотрудничество и улучшение ядерной безопасности в Болгарии

Болгария всегда была открыта для международного сотрудничества и обмена опытом. По различным причинам и, прежде всего, из-за разделения мира на блоки в недалеком прошлом, значительная часть нашего международного сотрудничества проводилась в рамках бывшего СЭВ и в очень малой степени с некоторыми западными странами. После 1990 г. и, особенно, после серьезных возражений против некоторых проектных недостатков и, прежде всего, ухудшенной эксплуатации АЭС с реакторами Советского производства, были установлены новые взаимоотношения в области международного сотрудничества. Более того, устранение блокового противостояния устранило и ряд других, в первую очередь, бюрократических барьеров и открыло новые возможнсти для международного сотрудничества в ядерной области.

Как уже было сказано в самом начале, мы оказались свидетелями настоящей интернационализации проблем безопасности и эксплуатации АЭС. Это сотрудничество осуществляется как под эгидой международных организаций, таких, как IAEA, WANO, OECD, NEA, так и на двусторонней основе с США, Японией, Германией, Бельгией, Великобританией и особенно с Францией.

В рамках группы 24-х, Болгария является объектом особенного внимания и помощи. Только за последние три года для Болгарии были спроектированы и финансированы более 81 проекта общей стоимостью свыше 64-х миллионов долларов.

Мы являемся членами и получаем информацию в наиболее значимых для АЭС проектах таких, как проект ACE (Advanced Containment Experiments), проект IPIRG (International Piping Integrity Research Group) и др. Болгария получает безвозмездное право пользования практически всеми кодами, разработанными для NRC, связанными с оценкой безопасности.

Наши специалисты - желанные консультанты в МААЕ, ВАНО и др.

У нас иногда с неуважением говорят о программах помощи по линии ФАР, ЕИО и др. Очевидно, это высказывания, демонстрирующие непонимание принципов ядерной безопасности и, прежде всего, понятия "культура безопасности". С увереностью можно сказать, что мировоззрение болгарских ядерных специалистов сейчас простирается дальше границ бывшего СССР и СЭВ. Кроме того, в этот момент мы знаем, что ядерная энергетика и ядерные энергетики Болгарии - это часть мирового ядерного потенциала, и что мы не одни при решении наших проблем, насколько трудными они бы не были. Это основной результат международного сотрудничества, и он должен быть правильно оценен.

В заключение можно сделать несколько важных выводов из всей предшествующей работы и особенно трудных последних лет.

Болгария и конкретно АЭС "Козлодуй" сделали важный шаг в создании нового типа культуры безпасности, основанного на глубокой переоценке уровня безопасности действующих блоков и полной прозрачности действий и намерений в национальном и международном планах. Определенный вклад в это имеет и КИАЭМЦ и особенно Инспекция по безопасному использованию атомной энергии. Мы готовы обсуждать наши проблемы со всеми, кто приходит с добрыми намерениями. Не все те, которые посещали нас в последние годы , были такими.

Будущее ядерной энергетической программы Болгарии и, в первую очередь, ее безопасность будет зависеть от правильного решения некоторых важных проблем:

1. На первом месте - успешное завершение программы безопасности, реконструкции и модернизации АЭС "Козлодуй". В то же время мы должны убедить наше общество, что атомная энергия - это экологически самый чистый в стране источник электроэнергии, который может быть дешевым, безопасным и надежным.
2. На втором месте: экономическое развитие страны определит будущую потребность от новых мощностей, в том числе и ядерных. Мы должны знать, что электроэнергия будет дорогостоящей на Балканах и в большой степени определит возможность ее развития.
3. Для болгарских ядерных специалистов и особенно для специалистов, эксплуатирующих АЭС "Козлодуй" нет дороги назад. Безопасность сооружений - это задача первостепенной важности, и начавшийся процесс перемен в этом отношении не должен прерываться.
4. И, наконец, мы должны развивать нашу собственную инфраструктуру инженерных, консультантских, научных и производственных организаций, что вляется гарантией безопасности и надежной работы наших АЭС.

При решении всех этих проблем одной из самых тяжелых задач будет убеждать, часто меняющуюся государственную администрацию и политиков в необходимости того, что мы делаем, и это будет вызовом для каждого из нас, но у нас нет другого выхода.

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PURPOSE

- COMPLY WITH INTERNATIONAL STANDARDS
- INTEGRATE INTO PLANT INFORMATIONNETWORK
- REDUCE OPERATION & MAINTEN ANCE COST
- DELETE OBSOLETE COMPONENTS
- DECREASE NUMBER OF DETECTOR CHANNELS

IMPLEMENTATION

- VVER - 1000 DESIGN
- STANDARDS
- SYSTEM ARCITECTURE
- DISTRIBUTED PROCESSING UNIT (DPU)
- CENTRAL RADIATION PROCESSOR (CRP)
- LOCAL RADIATION PROCESSOR (LRP)
- VVER - 1000 SPECIFIC RADIATION MONITORING SYSTEMS

VVER - 1000 DESIGN

SYSTEMS

- ACCIDENT LOCALIZATION (CONTAINMENT)
- BALANCE OF PLANT

DETECTOR CHANNELS

Type	U.S.PWR[1]	Soviet PWR1
Gas	12	16
Iodine	1	4
Noble Gas	16	3
Particulate & Lodine	2	6
Liquid	16	17
Wide Range Gas	4	3
Area	138	237

NOTE[1] Typical Pressurized Water Reactors

STANDARDS

INTERNATIONAL

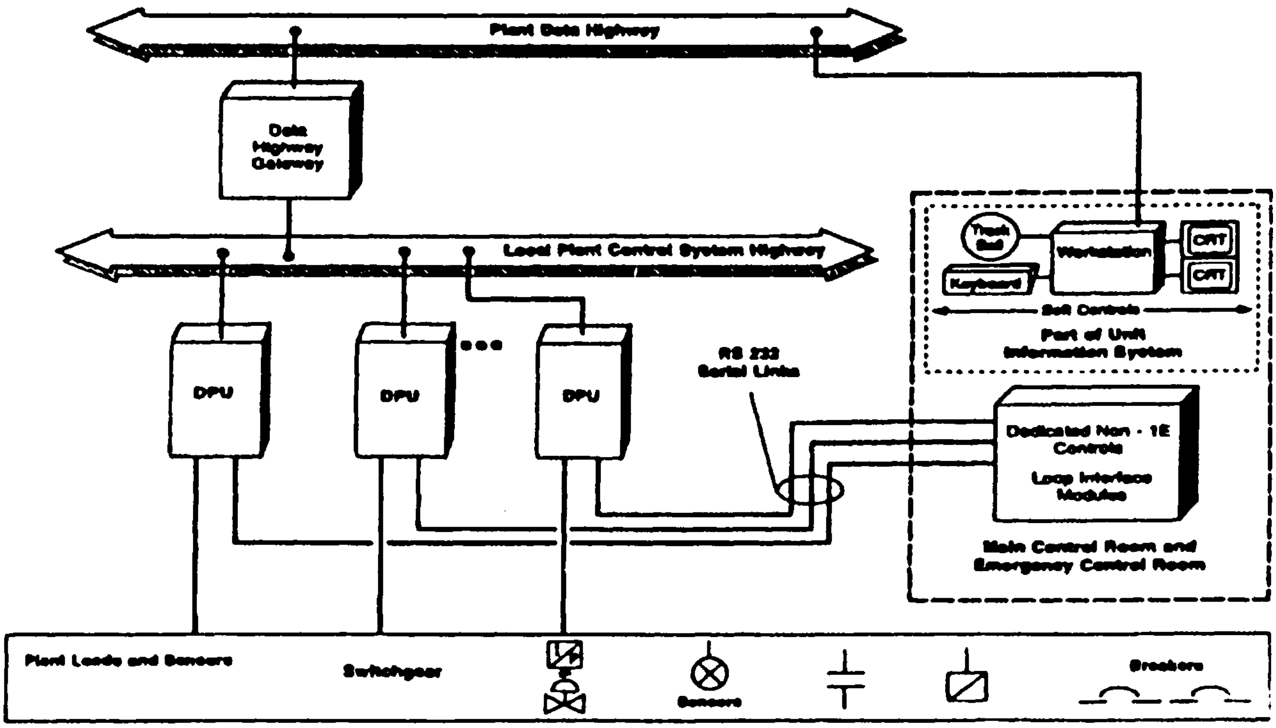
- INTERNATIONAL ELECTROTECHNICAL COMMISSION (IEC)
- INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS (IEEE)
- UNITED STATES NUCLEAR REGULATORY COMMISSION (USNRC)

NATIONAL

- NATIONAL REGULATORY AUTHORITY

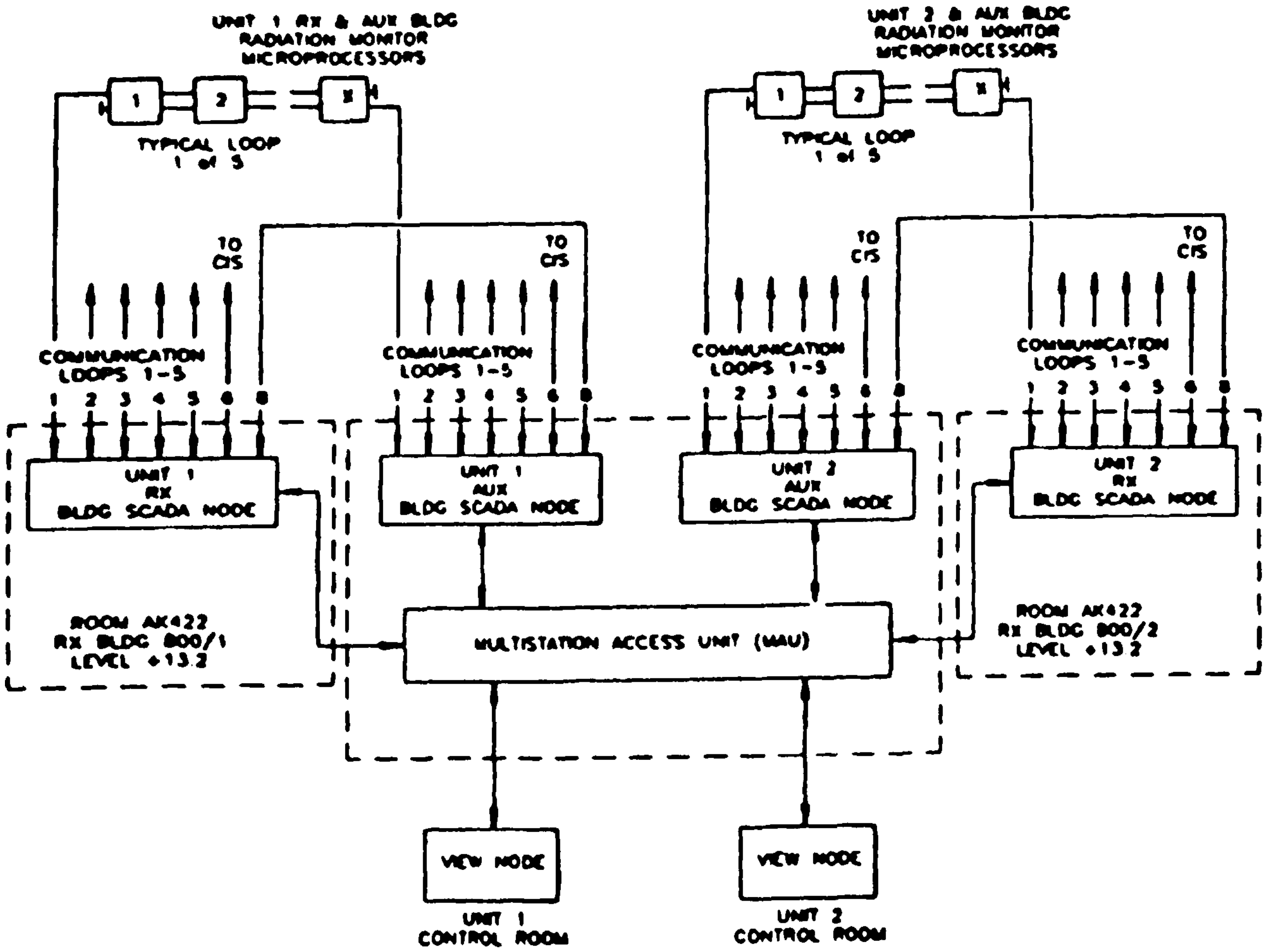
EQUIPMENT SPECIFICATION

I&C SISTEM ARCHITECTURE

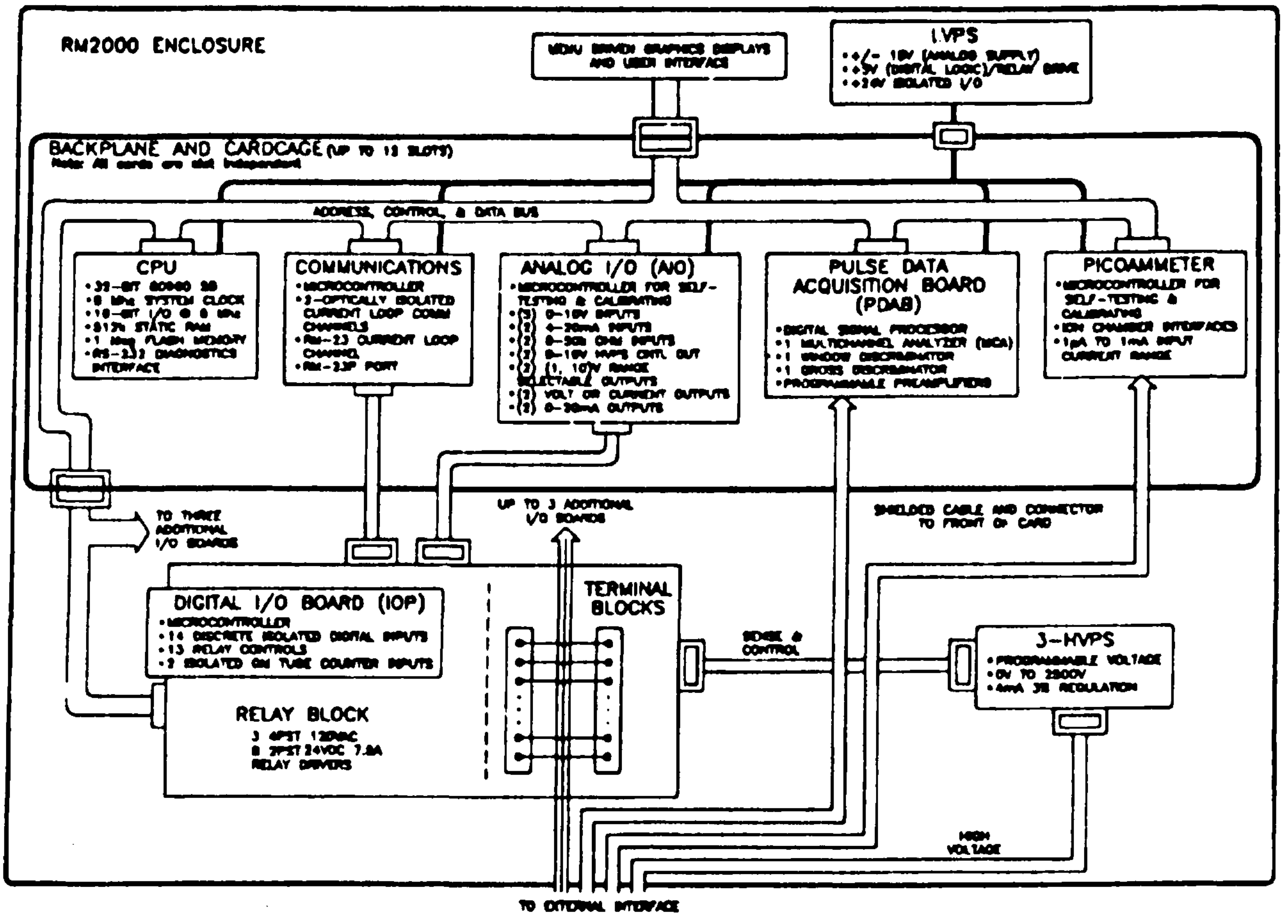


DPU = Distributed Processing Units

DRAMS ARCHITECTURE



LRP BLOCK DIAGRAM



VVER - 1000 SPECIFIC DESIGN

- REACTOR BUILDING STACK
- HERMETIC ZONE
- WASTE WATER BASIN
- DISTRICT HEATING

CONCLUSION

A DIGITAL RADIATION MONITORING SYSTEM (DRMS) WHICH MEETS INTERNATIONAL AND LOCAL REGULATIONS MAY BEINTEGRATED INTO A VVER - 1000 I&C MODERNIZATION PROGRAM. IF PLANNED AND IMPLEMENTED PROPERLY, THE I&C AND DRMS MODERNIZATION PROGRAM CAN PROVIDE COST SAVINGS BY REDUCING TIME REQUIRED TO ACCESS AND DISPLAY DATA AND REDUCE MAINTENANCE COST BY DELETING OBSOLETE PARTS AND DECREASING THE NUMBER OF DETECTOR CHANNELS.

ВОПРОСЫ БЕЗОПАСНОСТИ В ПРОЕКТАХ АЭС НОВОГО ПОКОЛЕНИЯ С ВВЭР

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Основные направления по дальнейшему совершенствованию реакторных установок ВВЭР:

1. Улучшение характеристик активных зон реакторов, включая характеристики повышающие самозащищенность реактора;
2. Увеличение еффективности механических органов управления и защиты реактора;
3. Более широкое использование пассивных систем безопасности для предотвращения тяжелых аварий и исключения радиоактивных выбросов свыше допустимых значений;
4. Обеспечение пассивного отвода остаточных тепловыделений от активной зоны при длительном полном обесточиваний блока, включая потерю источников надежного электропитания переменного тока;
5. Повышение надежности барьеров на пути выхода активности (оболочка твэл, границы давления первого контура, защитная оболочка);
6. Повышение надежности работы систем, важных для безопасности;
7. Широкое использование систем диагностики и оснащение блоков современными системами АСУТП.

Основные направления разработки реакторых установок ВВЭР для строящихся и проектируемых энергоблоков

Реакторная установка повышенной безопасности с реактором ВВЭР-1000 (В-392)

Создание реакторной установки повышенной безопасности с реактором ВВЭР-1000 (рис.1) на базе опыта, накопленного при разработке и эксплуатации базового проекта ВВЭР-1000 (В-320), является эволюционным путем совершенствования энергоблоков с реакторами ВВЭР по пути повышения безопасности ядерной энергетики.

Повышение безопасности РУ при проектировании достигается модернизацией реактора, применение дополнительных систем безопасности и усовершенствованием компоновочных решений, в основу которых положены следующие решения, присущие данной реакторной установке:

- реактор (по сравнению с реактором РУВ-320) модернизован за счет увеличения количества органов СУЗ с 63 до 121 шт., что позволяет обеспечить перевод реактора в подкритическое состояние при t=100 - 120 °С без дополнительного ввода борного раствора; применения активной зоны трехгодичной кампании с выгорающими поглотителями (на 1 этапе применяются борные СВП, в последующем уран-гадолиниевое топливо) с обеспечением отрицательного коэффициента реактивности по температуре теплоносителя во всех эксплуатационных состояниях реактора;

- применение систем диагностики для периодического контроля оборудования на остановленном реакторе и систем оперативной диагностики на работающем реакторе;

- применение главного циркуляционного насоса с негорючей смазкой и с конструкцией уплотнений, исключающей их течи при длительном полном обесточивании АЭС, что позволяет сохранить плотность ГЦК при отсутствии подачи каких либо охлаждающих сред;

- создание автоматизированной системы контроля текущего состояния и готовности систем безопасности к выполнению свох функций позволяет своевременно выявить отказы в системах безопасности и принять необходимые решения;

- разработка системы защиты оболочки от привышения давления выше допустимого за счет системы пассивного дожигания водорода с очисткой и сбросом парогазовой среды через специальные фильтры высокой эффективности;

- вынос резервного щита управления за пределы главного корпуса, а также создание общестанционнго аварийного пункта для контроля за состоянием аварийного энергоблока после тяжелой аварии;

- усовершенствование компоновки ГЦК (понижение отметки под установку оборудования) и применение двойной герметичной оболочки;

- применение пассивной системы отвода остаточных тепловыделений (СПОТ). Проектная система СПОТ состоит в том бы, что в случае полного обесточивания АЭС, включая потерю аварийного электроснабжения, был обеспечен отвод остаточного тепла без повреждения активной зоны реактора и границы давления 1 контура. В СПОТ используется воздушный теплообменник, установленный вне контаймента станции. Теплообменник соединен со вторым контуром парогенератора по пару и воде так, что пар из парогенераторов конденсируется в топлообменнике, отдав свое тепло наружному воздуху, и конденсат возвращается в водяной объем парогенератора;

- применяемые в системе аварийного охлаждения активной зоны двух ступеней гидроемкостей с давлением 6,0 МПа и 1,2 МПа. При авариях с течью I контура с наложением обесточивания АЭС, включая потерю аварийного электропитания, на первой стадии срабатывают гидроемкости с давлением 6,0 МПа, на последующей стадии развитие аварии срабатывают гидроемкости с давлением 1,2 МПа. Таким образом некоторое время без работы насосов системы САОЗ обеспечивается поддержание активной зоны в состоянии, при котором она залита водой и выполняются критерии по топливу;

- применение системы быстрого ввода бора в теплоноситель 1 контура позволяет заглушить реактор в запроектном режиме отказа механической системы аварийной защиты реактора(режим ATWS). Система представляет собой гидроемкость с запасом боронго раствора высокой концентрации и трубопроводов соединяющих ее с 1 контуром. При нормальной работе РУ емкость отключена от ГЦК. В случае, если произошло событие, требующее аварийной остановки реактора, но эта функция не была выполнена системой твердых поглотителей, емкость подключается к петле. При этом борный раствор за счет напора ГЦН поступает в реактор и происходит гашение ядерной реакции активной зоны реактора;

- в техническом обосновании безопасности рассмотрен более широкий спектр аварийных режимов. Введены запроектные тяжелые аварии с проведением вероятностных оценок безопасности.

Реакторная установка нового поколения для энергоблоков электрической мощностью 1000-1100 МВт (ВВЭР - 1100НП).

К основному отличию концепции данной реакторной установки от проекта В-392 следует отнести более широкое использование пассивных принципов при построении систем безопасности. Принципиальная схема этих систем показана на рис. 2

Системы состоят из двух ступеней гидроемкостей и пароводяных воздушных или иного типа теплообменников.

В состав систем также входит основное оборудование 1 контура (корпус реактора, трубопроводы и парогенераторы).

Системы предназначены для аварийного охлаждения и отвода остаточных тепловыделений реактора в проектных авариях с потерей активных систем расхолаживания. Пассивный характер работы системы позволяет использовать системы при полном отказе всех источников энергоснабжения.

Основные режимы работы систем:

а) длительное отсутствие всех источников энергоснабжения, включая дизельгенераторы, при сохранении плотности основных технологических контуров.

Отвод тепла от активной зоны реактора в данном режиме осуществляется за счет двух контуров и естественной циркуляции:

- водоводяной циркуляции теплоносителя 1 контура, организуемой за счет разницы высот между активной зоной теплопередающей поверхностью парогенераторов;
- пароводяной циркуляции теплоносителя в теплообменнике со сливом конденсата в объем котловой воды;

б) длительное отсутствие всех источников энергоснабжения, включая дизельгенераторы с разуплотнением 1 контура, вплоть до разрыва главного циркулационного трубопровода.

Обеспечение безопасности и функционирование реакторной установки предполагается осуществить за счет следующих основных положений концепция АЭС, оказывающих определяющее влияние на конструктивные решения по реакторной установке и системам:

- повышение параметров реакторной установки для повышения экономической эффективности энергоблока за счет увеличения длины твэл и выравнивания поля энерговыделения по объему активной зоны;
- совершенствование активной зоны и улучшение характеристик внутренней самозащищенности за счет:

использования выгорающего поглотителя, включаемого непосредственно в топливо;

уменшения поглащения нейтронов конструкционными материалами;

улучшения маневренных характеристик топлива;

повышения еффективности аварийной защиты для обеспечения снижения мощности и расхолаживания до 100 °С без ввода борного раствора в теплоносител;

- повышение ресурса работы основного оборудования до 50-60 лет;
- применение в проекте концепции "Течь перед разрывом" при проектировании опорных конструкций оборудования и трубопроводив и определении нагрузок на внутри-корпусные устройства оборудования при проектных авариях разрыва трубопроводов первого и второго контуров;
- совмещение активных систем аварийного охлаждения с системами нормальной эксплуатации, при этом требования к быстродействию систем, вытекающие из необходимости обеспечения охлаждения при авариях, будут

снижены в связи с повышением роли пассивных систем при обеспечении безопасности;

- повышение роли пассивных систем при обеспечении безопасности за счет:

создания контура естественной циркуляции для отвода тепла от парогенераторов и конечному поглотителю тепла. В качестве конечного поглотителя тепла используется атмосферный воздух или водяные объемы расчитанные на выкипание;

разработки подсистемы гидроаккумуляторов, находящихся под давлением газа различного давления вплоть до атмосферного, последовательно включающихся в работу.

- применение специальной системы быстрого ввода бора в дополнение к механической системе аварийной защиты реактора для обеспечения снижения мощности реактора при авариях с несрабатыванием аварийной защиты (ATWS);

- создание системы улавливания и охлаждения расплавленной активной зоны за пределами корпуса реактора.

Основные параметры реакторной установки нового поколения в сравнении с параметрами реакторной установки повышенной безопасности (проект В-392) представлены в таблице 1.

Реакторная установка нового поколения для энергоблоков электрической мощностью 500-600 МВт (ВВЭР 500/600)

Повышение безопасности АЭС осуществляется также за счет преимущественного использования пассивных систем безопасности для аварийного охлаждзения активной зоны и отвода остаточных тепловыделении от реактора:

- пассивная система САОЗ обеспечивает подачу воды в реактор из гидроемкостей и емкостей с большим аварийным запасом воды и далее слив в басейн отвода остаточных тепловыделений. Организация вокруг корпуса реактора бассейн отвода остаточных тепловыделений позволяет в случае разгерметизации 1 контура сохранить активную зону под уравнем воды, исключить ее перегрев за счет отвода тепла от стенок оборудования и за счет создания контура естественной циркуляции;

- пассивная система отвода тепла от герметического объема обеспечивает на первом этапе аварии при высоком уровне остаточных тепловыделений отвод тепла са счет испарения части воды в специальных баках. При снижении уровня остаточных тепловыделений через несколько суток тепло из гермообъема передается через стальную оболочку к конечному поглотителю - воздуху;

- система пассивного отвода тепла через 2 контур обеспечивает отвод тепла от РУ при герметичном 1 контуре. Принципиальная схема систем безопасности показана на рис. 3.

Максимально используются технические решения потвержденные опытом эксплуатации ВВЭР-1000 и ВВЭР-440.

Основные технические характеристики 4-х петельного варианта представлены в табл.1.

Размеры корпуса реактора ВВЭР-500/600 приняты такими же, как и для ВВЭР-1000. Основные технические решения по парогенератору аналогичны решениям для ПГВ-440, надежность которых подтвержена длительным опытом эксплуатации.

При этом:

- средняя удельная энергонапряженность топлива 55-65кВт/л вдвое меньше, чем в ВВЭР-1000;

- увеличен удельный объем воды в 1 контуре, в том числе и за счет использования компенсатора давления для ВВЭР-1000;

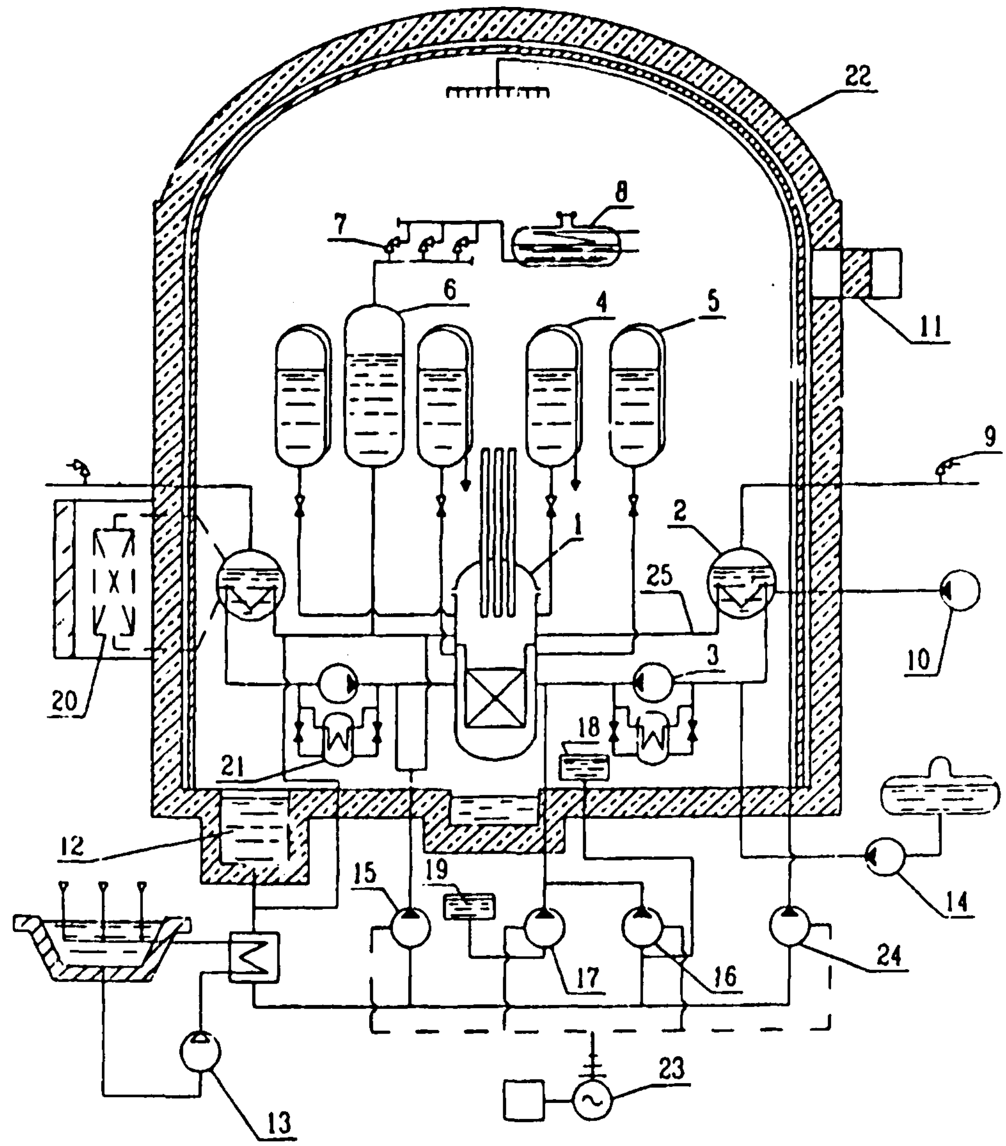
- снижен флюенс на корпус реактора.

Эффективность механической системы органов регулирования исключает несанкционированный выход реактора в критическое состояние при аварийном охлаждении активной зоны до температуры 100 °С с учетом застревания сомой эффективной группы СУЗ. Дополнительно предполагается, что при этом происходит полное замещение борной кислоты 1 контура чистым конденсатом из-за отказа соответствующих систем или в результате диверсии. Эта безопасность обеспечивается в любой момент работы топливной загрузки, даже сразу после перегрузки, когда в реактор загружено свежее топливо. Также как и в проектах, базирующихся на ВВЭР-1000 используются выгорающие поглотители. Это позволяет иметь низкую концентрацию борной кислоты в начале загрузки, что усиливает внутренние отрицательные обратные связи, приводящие к самогашению цепной реакции при увеличении мощности и температуры теплоносителя 1 контура.

Таблица 1

Наименование характеристики	ВВЭР-440	ВВЭР-1000 (В-392)	ВВЭР-500/600	ВВЭР 1100 НП
Мощность тепловая номинальная	1375	3000	1800	3300
Компоновка I контура	петлевая	петлевая	петлевая	петлевая
Число петель	6	4	4	4
Паропроизводительность в номинальном режиме, кг/с	753	1633	970	1600
Расход теплоносителя через реактор в номинальном режиме (по холодным ниткам), $м^3/с$	11,6	23,0	14,26	23,6
Давление номинальное стационарного режима на выходе из активной зоны, МПа	12,4	15,7	15,7	15,7
Температура теплоносителя в реакторе, °С				
на входе	268	290	296	296
на выходе	297	320	327	330
Давление генерируемого насыщенного пара в парогенераторе при номинальной нагрузке, МРа	4,61	6,27	7,061	7,35
Время нахождения (кампания) топлива в активной зоне, год	3-4	3	5-6	3-5
Количество ТВС в зоне, шт				
кассеты ВВЭР-440	349			
кассеты ВВЭР-1000		163	163	163

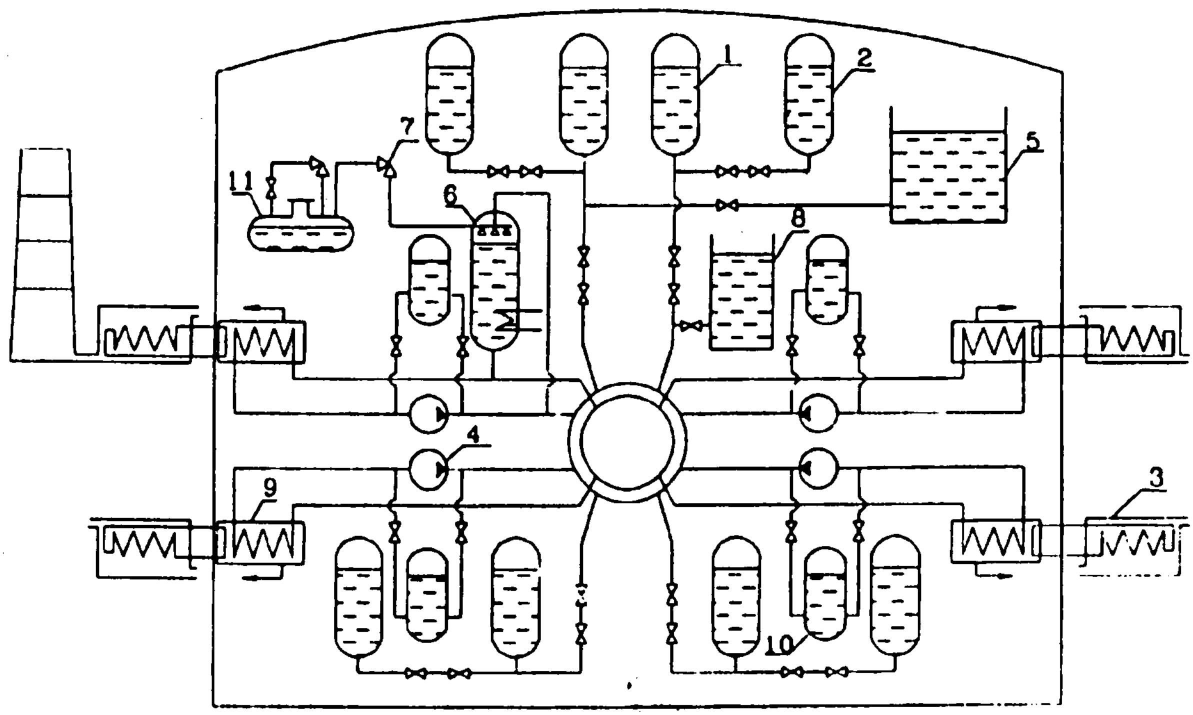
Принципиальная схема ВВЭР-1000 повышенной безопасности и дополнительных систем безопасности



1 - реактор
2 - парогенератор
3 - главный циркуляционный насос
4 - гидроемкость САОЗ P=0,6 МПа
5 - гидроемкость САОЗ P=1,2 МПа
6 - компенсатор давления
7 - предохранительный клапан КД
8 - барботер
9 - предохранительный клапан ПГ
10 - насос аварийной подпитки парогенератора
11 - фильтр на ГО
12 - бак-приямок с запасом борного раствора
13 - насос технической воды
14 - насос системы подпитки I контура
15 - насос расхолаживания системы САОЗ
16, 17 - насосы аварийного впрыска борного раствора высокого давления
18, 19 - баки борного раствора
20 - пассивная система отвода остаточного тепла
21 - пассивная система быстрого ввода бора
22 - защитная оболочка двойная
23 - дизель-генератор
24 - насос спринклерной системы
25 - главный циркуляционный трубопровод

Рис.1

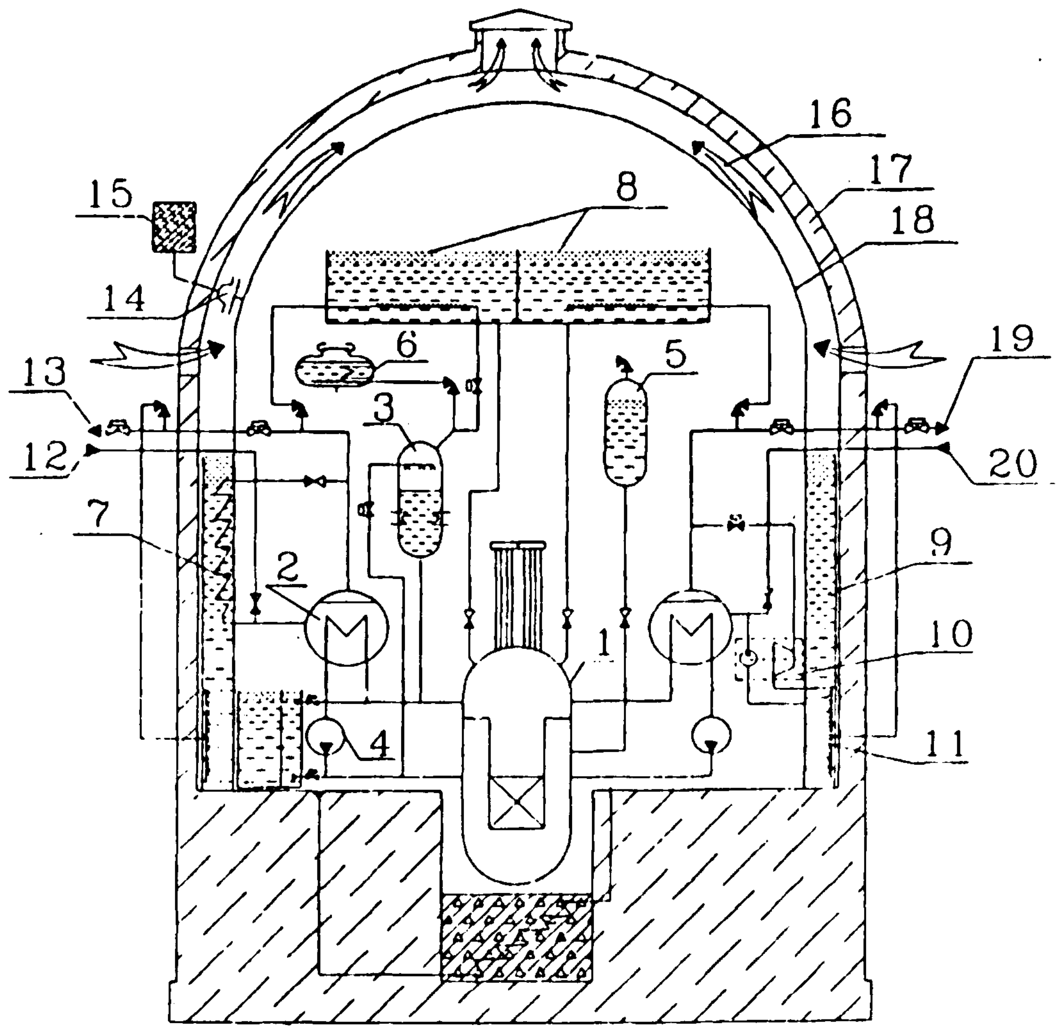
Принципиальная схема ВВЭР-1100 НП с дополнительными системами



1 – гидроемкость
2 – дополнительные гидроемкости
3 – система пассивного отвода тепла
4 – главный циркуляционный насос
5 – бак низкого давления
6 – компенсатор давления
7 – система сброса давления
8 – бак бассейна выдержки топлива
9 – парогенератор
10 – система ввода бора
11 – барботер

Рис. 2

РЕАКТОРНАЯ УСТАНОВКА ВВЭР-500/600
КОНЦЕПЦИЯ СИСТЕМ БЕЗОПАСНОСТИ



1. Реактор
2. Парогенератор
3. Компенсатор объема
4. ГЦН
5. Гидроемкость САОЗ
6. Барботер
7. Теплообменник СПОТ
8. Емкости САОЗ
9. Бак запаса ХОВ
10. Аварийный турбонасос
11. Барботажное устройство
12. Питательная вода
13. Пар на турбину
14. Мембрана
15. Фильтр
16. Охлаждение герметичной оболочки воздухом в аварийных режимах
17. Защитная оболочка
18. Герметичная оболочка
19. Пар на турбину
20. Питательная вода

Рис. 3

АЭС С ВВЭР - 1000: НАСТОЯЩЕЕ И БУДУЩЕЕ

В. Л. Молчанов, Г. И. Бирюков, В. П. Новак

АННОТАЦИЯ

19 энергоблоков с реакторной установкой ВВЭР -1000 четырех типов В-187, В-302, В-338, В-320 экспуатируются в России, на Украине и в Болгарии. Разработка проектов РУ осуществлялась в период 1971-1980 г. по нормативам, действовавшим в тот период. Введение новых нормативов, опыт эксплуатации отечественых и зарубежных АЭС требует внесения изменений и усовершенствования действующих АЭС, направления усовершенствований АЭС - повышение безопасности, надежности и экономичности.

1. ВВЕДЕНИЕ

В настоящее время в России, на Украине и в Болгарии энергоблоки АЭС с реакторами ВВЭР-1000 обеспечивают значительную долю обеспечения электроэнергией потребностей этих стран. Успешная эксплуатация АЭС, обеспечение их безопасности и надеждности является актуальной задачей. Проекты энергоблоков разрабатывались на основе нормативов, дествовавших в период 1970-1980-х годов, это:

- "Общие положения обеспечения безопасности атомных электростанций при проектировании, строительстве и эксплуатаций" (ОПБ-73);
- "Правила ядерной безопасности атомных электростанций" (ПБЯ-04-74).

Основным принципом безопасности, заложенным в проект АЭС с ВВЭР-1000, было обеспечение безопасности во всех проектых режимах, включая нормальные условия эксплуатации, нарушение нормальных условий и проектные аварии, с учетом наложения на исходное событие возможного единичного отказа любого элемента в системах важных для безопасности. Оценка уровня безопасности осуществлялась на основе детерминистического подхода. Проекты АЭС с ВВЭР-1000 были разработаны до того, как произошли две крупнейшие аварии на АЭС TMI и Чернобыльской АЭС, которые значительно изменили подходы и принципы обеспечения безопасности.

Опыт эксплуатации энергоблоков с ВВЭР-1000 выявил целый ряд недостатков, которые не позволяют достичь мировой уровень показателей экономичности работы энергоблока, в первую очередь по коэффициенту использования установленной мощности, трудно и дозозатратом на техническое обслуживание и ремонт.

Все это определило два основных направление усовершенствования дествующих энергоблоков АЭС с ВВЭР-1000:

- повышение безопасности;
- повышение надежности, эксплуатационной готовности и экономических показателей.

Кроме того, в настоящее время в России ведутся разработки новых проектов АЭС с реакторными установками ВВЭР-1000 на базе требований современных российских нормативов и с учетом накопленного отечественного и мирогого опыта проектирования и эксплуатации АЭС с ВВЭР (PWR). Развитие этих проектов идет в двух направлениях:

- эволюционном (концепция АС-91);
- с внедрением новых перспективных решений, основанных на принципах пасивносит и внутренней самозащищенности (концепция АС-2).

2. ОСНОВНЫЕ ПОДХОДЫ К ПОВЫШЕНИЮ УРОВНЯ БЕЗОПАСНОСТИ ДЕЙСТВУЮЩИХ АЭС

В настоящее время усилиями института "Атомэнергопроект" (Москва). ОКБ "Гидропресс", РНЦ "Курчатовский институт" разработат документ "Повышение безопасности действующих блоков атмных станций с ВВЭР-1000". Этот документ содержит общую концепцию повышиния безопасности эксплуатируемых в настоящее время энергоблоков, направленную на снижение риска населения и персонала за счет снижения вероятности возникновения исходных событий аварий и отслабления последствий аварий путем повышения барьеров на пути распространения радиоактивных веществ. При разработке концепции использовался подход, основанный на экспертной оценке степени влияния на глубоко эшелонированную защиту имевших место отступлений от современных НТД. За основу были приняты перечни отступлений от современных НТД, представленных в составе технических обоснований безопасности (ТОБ) действующих АЭС с ВВЭР-1000 и содержащие экспертную оценку возможных последствий имевшихся отступлений.

В качестве методологической основы была принята классификация МАГАТЭ по степени влияния указанных выше отступлений на глубоко эшелонированную защиту, разработанная миссией МАГАТЭ при анализе безопасности АЭС с ВВЭР-440 (РУ В-79, 230). При этом были выработаны 4 категории значимости имеющегося дефецита безопасности и срочности устранения этого дефицита.

Категория 1 - отступления от признанной международной практики.

Категория 2 - отступления, значимые для безопасности.

Категория 3 - отступления, имеющие высокую значимость для безопасности. Глубоко эшелонированная защита недостаточна. Требуются немедленные корректирующие действия.

Категория 4 - отступления, имеющие наибольшее значения для безопасности. Глубоко эшелонированная защита непрпиемлема. Требуются безотлагательные меры.

В резултате были сформулированы рекомендации по мероприятиям, направленным на повышение уровня безопасности действующих энергоблоков.

Мероприятия, отнесенные к 3 и 4 категории должны внедрятся в первую очередь. Общий срок реализации всей программы 4-5 лет.

Указанные мероприятия направлены на:

- предупреждение возникновения исходных событий;
- ликвидации недостаточного или неприемлемаго уровня глубоко эшелонированной защиты;
- снижение риска повреждения активной зоны;
- внедрение мер по управлению запроектными авариями.

Внедрение мероприятий предполагается осуществлять в два этапа:

- первы предустматривает разработку и внедрение организованных и распорядительных документов по предупреждению возникновения исходных событий; компенсирующие мироприятия по уменшению последствий отступлений 4 категории; разработка рекомендаций по управлению запроектными авариями имеющимися системами;

- второй - внедрение остальных намеченных мероприятий.

3. ОСНОВНЫЕ МЕРОПРИЯТИЯ ПО ПОВЫШЕНИЮ БЕЗОПАСНОСТИ ДЕЙСТВУЮЩИХ ЭНЕРГОБЛСКОВ

Замена теплоизоляции на трубопроводах и оборудовании в предалах геометричной болочки;

- Внедрение дополнительных средств подпитки ПГ от надежных источников;
- Мероприятия, исключающие транспортировку тяжелых грузов над реактором или бассейном выдержки;
- Модернизация средств контроля и управления ВХР 1 контура;
- Модернизация оборудования второго контура с целью улучшения ВХР второго контура;
- Модернизация перегрузочной машины;
- Разработка и внедрение авотматизированной системы управления распределением энерговыдаления в активной зоне;
- Обеспечение подпитки бассейна выдержки при закрытии локализующей арматуры в системе охлаждения бассеина;
- Разработка и внедрение автоматизированной сисемы управления распределением энерговыделения в активной зоне;
- Обеспечение подпитки бассейна выдержки при закрытии локализующей арматуры в системе охлаждения бассейна;
- Разработка и внедрение програмного комплекса для расчетного анализа остаточного ресурса элементов оборудования РУ;
- Мероприятия по повышению радиационного ресурса корпуса реактора;
- Мероприятия по улучшению систем контроля и диагностики;
- Разработка дополнительных обосновывающих материалов:

* вероятный анализ безопасности;
* рассмотрение запроектных аварий;
* рассмотрение проектных аварий с изпользованием западных вычислительных кодов;
* атестация и верификация использованных в проекте вычислительных кодов;
* усовершенствование эксплуатационной документации на основе симптомно-ориентированного подхода, документации по техническому обслуживанию и ремонту.

4. МЕРОПРИЯТИЯ ПО ПОВЫШЕНИЮ ЭКСПЛУАТАЦИОННОЙ ГОТОВНОСТИ И ЭКОНОМИЧЕСКИХ ПОКАЗАТЕЛЕЙ ДЕЙСТВУЮЩИХ ЭНЕРГОБЛОКОВ

- Оптимизация топлинного цикла

- Разработка регламентов, инструкций и других документов по периодической порверке систем, важных дле эксплуатации, по техническому обслуживанию и ремонту;

- Повышение надежности уплотнения главного разъема реактора;

- Повышение надежности регулируемых клапанов уровня в ПГ, подогревателях высокого (ПДВ) и низкого давления (ПНД);

- Разработка спецоснастки для проведения технического обслуживания и ремонта с учетом снижения дозовых и трудовых затрат при ППР и др.

5. БУДУЩЕЕ АЭС С ВВЭР-1000

Развитие проекта РУ ВВЭР-1000 в России осуществляется в двух направлениях. Первое направление - постепенное, эволюционное совершенствование базового поректа РУ ВВЭР-1000 (В-320) путем внедрения в него новых усовершенствований, отработка которых возможна експериментальным, расчетным путем без создания крупномасштабных или натурных образцов. Указанное направление в настоящий момент представлено проектом АС-91.

Второе направление развития ВВЭР-1000 предусматривает внедрение новых решений для удовлетворения технических требовании для АЭС, пуск которых предполагается после 2000-2005 года. Предполагаемые конструктивные решения требуют для их всестороннего исследования создания крупномасштабных или натурных стендов. Данное направление представлено проектом АС-92.

5.1 КОНЦЕПЦИЯ АС-91

Концепция АС-91 разработана Санкт-Петербургским институтом "Атомэнергопроект" совместно с ОКБ "Гидропресс" и другими проектными организациями России с участием фирмы ИВО Интернешнл (Финляндия).

Проект базируется на российских нормах и стандартах. Наряду с российскими требованиями проект отвечает требованиям гайдов, изданных финскими властями в области ядерной безопасности.

параметры, харктеризующие теплогидравлическую обстановку в реакторе, близки к проектным для серийной реакторной установки В-320, которая имеет большой опыт эксплуатации.

Основный принцип, заложенный в концепцию проекта АС 91 - еволюционный характер внасимых модификаций, не требующих проведения натурных или крупномасштабных испытаний для доказательств их применимости:

- учет мирового и российского опыта эксплуатации, внедрение передовых апробированных технологий;

- оптимизация затрат трудовых, материальных, дозовых с целью создания блока, конкурентноспособного на мировых ранках по экономическим показателям;

- учет и удовлетворение международным и российским нормативам.

В проекте применено и реализовано следующее.

По безопасности:

- Усовершенствованныи реактор ВВЭР-1000.
- Усовершенствованный парогенератор ПГВ-1000У;
- Принцип четырех каналов систем безопасности;
- Усовершенствованный главый циркуляционный насос ЦНА-1391,
- Двоиная защитная оболочка;
- Системы сброса давления и очистки выбросов из защитнои оболочки;
- Дополнительные системы безопасности: системы возврата теплоносителя первого контура при течи из первого контура во второй и система впрыска бора высокого давления;
- Улучшенная компоновка реакторного отделения;
- Управление тяжелыми, запроектными авариями, включая маловероятные, приводящие к расплавлению активнои зоны.

По технико-экономическим показателям:

- Усовершенствованный топливныи цикл: использование тепловыделяющих сборок, внедрение перегрузок типа "in-in-out" и другое;
- Увеличенный срок службы основного оборудования с 30 до 40 лет;
- Концепции "течь перед разрушением".

Применение усовершенствованного реактора ВВЭР-1000 позволяет повысить еффективность органов авариинои защиты реактора с целью поддержания реактора в подкритическом состоянии при расхолаживании до температуры 100 °С без ввода борного раствора.

Предусмотрено улучшение обратных связей по реактивности: за счет обеспечения отрицательных коэффициентов реактивности по температуре теплоносителя и топлива в течение всей компании исключается опасное повышение давления в первом контуре в режимах с отказом на срабатывание аварийной защиты.

Проект АС-91 выбран КНР в качестве проекта, которыи должен реализовываться в провинции Ляонин по межправительственному соглашению между РФ и КНР. Рассматривается также возможность реализации его на энергоблоке No4 Юу АЭС на Украине.

5.2 КОНЦЕПЦИЯ АС - 92

Концепция АС-92 разработана Московским институтом "Атомэнергопроект" совместно с ОКБ "Гидропресс" и другими проектными организациями России .

Проект базируется на российских нормативах.

Оборудование реакторной установки практически не отличается от применяемого в проекте РУ для АС-91.

Это направление предполагает:

- Учет с использование лучшихрешений проекта АС-91;
- Внедрение пассивных систем безопасности в сочетании с активными;
- Совершенствование топливного цикла (маневренное топливо, разработка уран-плутониего цикла и т.д.).

Основное принципиальное отличие этой концепции заключается в применении пассивных систем безопасности.

В этом проекте, дополнительно к имеющимся активным системам безопасности, для исклЮчения повреждение топлива в тяжелых авариях, предусматривается введение дополнительных пассивных систем.

BG9600402

ЭНЕРГОБЛОКИ АЭС ПЕРВОГО ПОКОЛЕНИЯ С РЕАКТОРНЫМИ УСТАНОВКАМИ ВВЭР-440 (В-230)

Вопросы безопасности и модернизации АЭС с ВВЭР-440 (В-230)

Н. Мельников

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1. Введение

За десятилетний период с 1971 по 1981 г. в России, Армении и странах Восточной Европы были построены и введены в эксплуатацию 16 энергоблоков АЭС первого поколения с реакторами ВВЭР-440 (см. таблицу)

Таблица

Страна	Название АЭС	Номер блока	Год пуска блока	Проектный срок службы, лет	Тип реакторной установки
1	2	3	4	5	6
Россия	Нововоронежская АЭС	3 4	1971 1972	30 30	В-179 В-179
	Кольская АЭС	1 2	1973 1974	30 30	В-230 В-230
Армения	Армянская АЭС	1 2	1976 1980	30 30	В-270 В-270
Германия	АЭС Норд	1 2 3 4	1973 1974 1977 1979	30 30 30 30	В-230 В-230 В-230 В-230
Болгария	АЭС Козлодуй	1 2 3 4	1974 1975 1980 1981	30 30 30 30	В-230 В-230 В-230 В-230
Словакия	АЭС "Богунице"	1 2	1979 1980	30 30	В-230 В-230

В настоящее время в России, Болгарии и Словакии в эксплуатации находятся 10 энергоблоков этой серии. Это реальность, с которой нельзя не считаться, имея ввиду необходимость обеспечения условий их безопасной эксплуатации.

В Западных странах, а также в странах, в которых эксплуатируются АЭС первого поколения по праву имеет место озабоченность безопасностью атомных электростанций, построенных по проектам бывшего Советского Союза. Требования со стороны Запада остановить эти АЭС таят в себе желание очень просто решить вопрос исключения риска. Однако, вопросы модернизации блоков или вывода их из эксплуатации не должны рассматриваться в отрыве от состояния альтернативных источников энергии и экономической ситуации в стране.

Характерным примером поспешного принятого решения о выводе из эксплуатации атомной электростанции этого типа является Армянская АЭС.

Целью настоящей работы является показать, что атомные электростанции с ВВЭР-440 (В-230) могут быть доведены в пределах расчетного их срока службы до приемлемого уровня безопасности за счет внедрения технических и организационных мероприятий, соответствующих современным подходам к обеспечению безопасной эксплуатации АЭС.

С другой стороны очевидно, что государства, в силу сложившихся условий, сами не в состоянии обеспечить своими силами все объемы работ по модернизации блоков без технической и экономической помощи Западных государств. Помощь, оказываемая по программам PHAPE и TACIS.

Чтобы реально улучшить условия безопасности на оставшийся срок эксплуатации этих энергоблоков.

2. Проектные основы и положительные особенности реакторных установок ВВЭР-440 первого поколения.

Несмотря на отсутствие в период создания проектов реакторных установок В-179, В-230 (конец 60-х годов) отечественной нормативно-технической документации по безопасности для атомных электростанций, наиболее ответственное оборудование (реактор, парогенератор, компоненты главного циркуляционного контура и др.) выполнялось по специально разработанным нормам и правилам, применяемым в военном судостроении. В частности, при изготовлении указанного оборудования были использованы "Основные положения" на сварку и наплавку и "Правила контроля", апробированные в судостроении, в том числе для реакторного оборудования транспортных установок. Эти нормы и правила явились хорошей основой для действующих в настоящее время нормативных документов при изготовлении и монтаже оборудования для АЭС.

Поэтому при создании первых АЭС были предусмотрены меры обеспечения качества конструирования, изготовления и монтажа оборудования, повышенных требований к эксплуатации. Основное оборудование реакторной установки и трубопроводы имели высокую степень надежности и соответствующие конструктивные запасы, которые практически исключали возможность крупных разрывов трубопроводов, что позволяло ограничить размер течи до Ду32. Для ограничения истечения трубопроводы диаметром 50, 70 и 100 мм, подсоединяемые к главному циркуляционному трубопроводу, снабжены вставками диаметром 32 мм, что обеспечивает максимальную расчетную величину истечения, эквивалентную отверстию диаметром 32 мм.

Трубопроводы главного циркуляционного контура, вспомогательных систем первого контура, задвижки, насосы и теплообменная поверхность парогенераторов выполнены из нержавеющих аустенитных сталей типа 0Х18Н10Т, не подверженных хрупким разрушениям.

При выборе максимальной проектной аварии это давало основание считать невозможным возникновение мгновенных разрушений крупных трубопроводов. Принцип невозможности возникновения мгновенных разрушений больших трубопроводов, работающих под давлением, сформулирован в действующей в настоящее время в ФРГ фундаментальной концепции безопасности (Basis Safety Concept). Многолетний опыт эксплуатации энергоблоков этого типа подтверждает надежность трубопроводов первого контура.

Оборудование и трубопроводы первого контура, работающие под давлением, не имеют продольных сварных швов, имеют наиболее простую конфигурацию, позволяющую наиболее достоверно определить напряженное состояние и тем самым подтвердить прочность для всех проектных режимов. Так, нижняя часть корпуса реактора, где размещается активная зона, выполнена в виде гладкой цельной цилиндрической оболочки с эллиптическим днищем, без каких-либо врезок и отверстий, что исключает дополнительные концентраторы напряжений. Патрубки корпуса реактора для приварки главных циркуляционных трубопроводов Ду500 мм выполняются как одно целое с обечайкой, что позволяет исключить сварные швы значительных размеров и большой протяженности.

Теплотехнические обоснования исходили из того, что при принятой максимальной проектной аварии уровень воды в реакторе всегда будет выше уровня активной зоны, подача воды насосами аварийной подпитки расходом 100 $м^3$/час обеспечивает достаточный теплосъем, исключающий повреждение тепловыделяющих элементов.

При разработке предложений по модернизации АЭС с ВВЭР-440 следует принимать во внимание также преимущества реакторной установки В-230, к которым относятся следующие:

1. Мощность источника радиационной опасности:

- реактор средней мощности 1375 Мвт (тепловых), соответствующий тенденции перспективных АЭС повышенной безопасности, предусматривающей применение принципов внутренней физической безопасности и пассивности.

2. Сопротивляемость развитию процессов с неуправляемым увеличением мощности:

- высокая эффективность органов защиты реактора;
- отрицательные коэффициенты реактивности по температуре теплоносителя во всех режимах работы;
- отрицательный паровой эффект реактивности;
- сильный отрицательный мощностной коэффициент реактивности;
- наличие системы температурного контроля реактора с большим количеством датчиков (контроль охватывает 60% сборок);
- устойчивость пространственного распределения энерговыделений.

3. Устойчивость 1 и 2 барьеров на пути распространения радиоактивных продуктов при нарушениях и авариях:

- максимальное значение линейной тепловой нагрузки на ТВЭЛ в 2,3 раза меньше допустимого и на 30% ниже, чем для реакторов типа PWR большей мощности;
- низкая объемная неравномерность тепловыделения (2.08);
- низкая неравномерность мощности отдельных ТВЭЛов (1.55);
- "устойчивая" оболочка ТВЭЛ;
- высокий запас по показателям теплотехнической надежности в исходном стационарном режиме (DNBR-4.0).

4. Устойчивость против возникновения неуправляемой потери теплоносителя первого контура:

- корпус реактора не имеет продольных швов;
- трубопроводы первого контура, главные запорные задвижки, главный циркуляционный насос, коллектор первого контура парогенератора, оборудование спецводоочистки выполнены из аустенитной стали, что создает предпосылки для применения принципа „течь перед разрушением";
- относительно большой запас теплоносителя первого контура над активной зоной, в системе компенсации давления, в петлях и собственный запас воды в зоне;
- минимальное количество присоединений к главному циркуляционному контуру в неотсекаемой части трубопровода;
- корпус реактора не имеет отверстий ниже главных патрубков;
- схема первого контура "без протечек", что обеспечивает высокую степень его герметичности, протечки теплоносителя приблизительно 0.5 $м^3$/сутки:

а) применены герметичные ГЦН типа ГЦН-310;

б) замкнутая система очистки воды первого контура;

в) фланцевые соединения имеют непрерывный контроль.

5. Схемные и конструктивные решения:

- многопетлевая схема отвода тепла от активной зоны, обеспечивающая высокую степень резервирования;
- наличие запорных задвижек на петлях, позволяющих отсечь поврежденный участок трубопровода;
- парогенераторы обладают большими запасами воды по второму контуру, позволяющие обеспечить отвод от активной зоны остаточного тепловыделения в течение 6-7 часов без подпитки;
- компоновка оборудования первого контура с применением горизонтального парогенератора обеспечивает теплоотвод от активной зоны с помощью естественной циркуляции теплоносителя после любого авариинoго переходного процесса, в том числе в процессе снижения уровней как в реакторе, так и по второму контуру в порагенераторе;

Приведенные характеристики активной зоны реактора и первого контура обеспечивают безопасность по отношению к аварийному отклонению в достаточно широком диапазоне основных параметров реакторной установки и отвечают ряду требований, предъявляемых к перспективным поколениям атомных электростанций.

Реальная эффективность положительных свойств реакторной установки с ВВЭР-440 проявилась во время инцидентов на Армянской АЭС и АЭС "Норд", связанных с пожаром в кабельных помещениях. При этом были потеряны все источники электропитания собственных нужд на время около 6-ти часов.

На Кольской АЭС ураганом были выведены все внешние источники электропитания, а на 2-х блоках из 4-х было потеряно питание собственных нужд II категории надежности (от дизельгенераторов) на время около 2-х часов.

В указанных аварийных ситуациях проблем с охлаждением реактора не возникло.

Важное преимущество ВВЭР-440 (В-230) с точки зрения безопасности - большую тепловую инерцию отмечает главный инспектор по ядерной безопасности фирмы "Электроситэ де Франс" г-н П.Танжи (см. журнал NEW 9-10/1993г.):

"Важность большой инерции ВВЭР-440 нельзя недооценить. Это хорошая особенность с позиции безопасности...

В подтверждение сказанному, финны, эксплуатирующие АЭС "Ловииза" (два блока с ВВЭР-440), утверждают, что блоки безопасны в течение 5-6 часов без каких-бы то ни было внешних источников энергии".

3. Основные направления повышения безопасности энергоблоков с ВВЭР-440 (В-179, В-230)

Рассматривая проблему повышения безопасности эксплуатации АЭС с реакторами первого поколения, в 1990 году в рамках WANO были разработаны "Предложения по реконструкции атомных станций типа ВВЭР-440 с реактором В-230" (Московский региональный центр).

В основу подхода было положено выявление дефицита безопасности и на основании ранее уже имеющихся материалов и предложений по реконструкции, с учетом "Позиции органов надзора по атомной энергетике..." стран-владельцев АЭС, были сформулированы конкретные технические мероприятия по следующим основным направлениям:

- расширение спектра проектных аварий;
- обеспечение целостности первого контура;
- повышение надежности компонентов систем, важных для безопасности;
- уменьшение вероятности возникновения исходных событий;
- повышение структурной надежности систем безопасности;

- повышение герметичности и целостности герметичных помещений, окружающих первый контур;

- улучшение противопожарной защиты;

- улучшение качества и эффективности системы радиационного контроля (внутреннего и внешнего);

- управление запроектными авариями;

- обеспечение сейсмостойкости АЭС.

По результатам работы миссий МАГАТЭ, относящихся к проекту и эксплуатационной безопасности в феврале 1991 года был выполнен концептуальный анализ имеющегося уровня безопасности АЭС с реакторами данного типа и выданы рекомендации концептуального характера. Рекомендации, связанные с вопросами проекта, могут быть приняты в качестве технической основы для выполнения модернизации, направленной на повышение безопасности эксплуатации АЭС.

В основе этого анализа была принята оценка степени влияния дефицитов безопасности на глубоко эшелонированную защиту. Были сформулированы 4 категории значимости имеющегося дефицита и рекомендованы сроки устранения этого дефицита.

Кроме того в анализе экспертов МАГАТЭ уделяется особое внимание вопросам эксплуатации (культуре безопасности), так как они затрагивают первые линии глубоко эшелонированной защиты, то есть обеспечению нормальных условий эксплуатации, поддержанию пределов и условий безопасной эксплуатации и предупреждению аварийных ситуаций на АЭС.

Указанные рекомендации должны найти отражение в программах модернизации энергоблоков АЭС с учетом особенностей блоков, особенностей организации их эксплуатации, экономических возможностей и других обстоятельств.

Необходимо, чтобы эксплуатирующие организации и регулирующие органы, ответственные за безопасность эксплуатации АЭС, одобрили программу модернизации и определили приоритеты технических мероприятий для реализации. В России в 1992-1993 г.г. для 1, 2 блоков Кольской АЭС и 3, 4 блоков Нововоронежской АЭС были разработаны "Концепции повышения безопасности АЭС..." и утверждены Концерном "Росэнергоатом". Концепции служили основой разработки графиков модернизации энергоблоков АЭС, разработки технических заданий на проектирование и т.д.

Для организации работ по модернизации энергоблоков АЭС разработаны графики для каждой АЭС и приказом министра атомной энергетики РФ введены в действие в 1993 году.

На основании выполненных за последнее время исследований и принятой концепции можно выделить следующие технические мероприятия по модернизации, необходимые для реализации:

1. Расширение спектра проектных аварий до течи первого контура с эквивалентным диаметром Ду100 мм включительно. При этом повреждение твэлов активной зоны не должно превышать пределы безопасной эксплуатации по количеству и величине дефектов твэлов, что составляет 1% твэлов с дефектами типа газовой неплотности и 0,1% твэлов, для которых имеет место прямой контакт теплоносителя и ядерного топлива.
Это предполагает выполнение необходимых расчетных анализов аварий.

2. Обеспечение целостности первого контура и контроля за его плотностью как одного из барьеров на пути распространения радиоактивных веществ.

Для чего необходимо:

- Обеспечить увеличенный обэем контроля металла оборудования и трубопроводов первого контура, включая контроль методом ультразвуковой дефектоскопии сварных швов трубопроводов различного диаметра из аустенитной стали;

- Обосновать применимость для трубопроводов первого контура каждого конкретного блока концепции "течь перед разрушением" и определить требования к режимам эксплуатации, техническому обслуживанию и эксплуатационному контролю.

Для обоснования принятого ограничения масштаба проектной аварии должно быть показано, что вероятность разрыва трубопровода с возникновением течи через эквивалентный диаметр более Ду100 мм менее 10^{-5} в год на блок с учетом фактических данных по конструкции и результатов изготовления, контроля металла при изготовлении и эксплуатации. Должно быть показано, что возможное разрушение трубопроводов реализуется только через этап "течь перед разрушением" с эквивалентным диаметром менее Ду100 мм, а система диагностики течи и режим эксплуатации могут обеспечить принятие необходимых мер до достижения сквозной трещины эквивалентного размера Ду100 мм.

3. Для внедрения концепции "течь перед разрушением" должно быть выполнено поэтапное оснащение блоков системами оперативной диагностики трубопроводов первого контура: диагностика течей, вибрации и обнаружения свободных предметов, контроль влажности в помещениях первого контура. Штатные системы контроля металла совместно с указанными системами диагностики позволят обеспечить обнаружение опасных дефектов, которые могут привести к течам с эквивалентным размером более Ду100 мм.

4. В обоснование радиационного ресурса корпусов реакторов выполнить для каждого блока расчет хрупкой прочности корпусов реакторов в эксплуатационных режимах с учетом фактического флюенса нейтронного потока и имеющихся в настоящее время данных по результатам исследований "лодочных" образцов, вырезанных из действующих корпусов.

Для восстановления пластических свойств металла сварного шва N4 и повышение эксплуатационной надежности корпусов были осуществлены некоторые меры, в том числе проведен отжиг ряда корпусов. Однако результаты исследований образцов, вырезанных из корпусов реакторов после отжига, не дают основания считать проблему хрупкой прочности окончательно решенной.

Разработанная имперwindow формула, позволяющая определить критическую температуру хрупкости T_{KO} исходя из содержания фосфора и меди, не дает вполне надежных результатов. Поэтому необходимо провести дополнительные исследования. Для этого предполагается выполнить отбор "лодочных" образцов из корпусов 3, 4 блоков Нововоронежской АЭС и 1, 2 блоков Кольской АЭС. Целью этих исследований является:

- проверить и расширить информацию о корреляции между результатами испытаний малогабаритных образцов и стандартных образцов Шарпи, а также уточнить корреляцию результатов с вязкостью разрушения K_{1C} на образцах из темплетов, вырезанных из корпуса реактора 2-го блока НВАЭС;
- расширить знания о повторном охрупчивании после отжига.

Проведение дополнительных исследований позволят уточнить эффективность процедуры отжига и критерии для определения допустимого интервала времени до следующего отжига корпуса реактора.

Для корпуса рактора 1-го блока АЭС "Козлодуи" с характерным для этого блока содержанием фосфора в сварном шве N4 целесообразно вырезать из сварного шва "лодочные" образцы для оценки изменения свойств такого металла при облучении и отжиге и корректировки существующей методики расчета.

Для этого корпуса ультразвуковой контроль должен проводиться не реже 1 раза в 4 года с высокой вероятностью (>0,9) выявления допустимых дефектов (расчетное значение не более 20 мм^2).

5. Внедрение системы внутриреакторного контроля (СВРК).

СВРК обеспечивает непрерывный и оперативняй сбор, обработку и предствление информации о параметрах первого и второго контура и о распределении энерговыделения по радиусу и высоте активной зоны.

Одновременно предусматривается модернизация каналов температурного контроля теплоносителя на выходе активной зоны для повышения их эксплуатационной надежности с заменой всех термопар на термопары ТХА-2076, установкой соединительных коробок КС-513М1 и изменением конструкции узла уплотнения выводов термопар.

6. Модернизация электрооборудования системы управления и защиты реактора в связи с недостатками существующей системы:

- моральный и физический износ электрооборудования и компонентов системы;

- отсутствие необходимого резервирования компонентов и разделение СУЗ реактора на независимые компоненты.

С целью устранения указанных недостатков предусматривается замена электрооборудования СУЗ на современный комплекс с введением 2-х независимых комплектов системы аварийной защиты по 3 каналам в каждом комплекте.

7. Модернизация аппаратуры контроля нейтронного потока (АКНП).

Аппаратуры контроля нейтронного потока имеет существенные недостатки, в том числе:

- моральный и физический износ линий связи и аппаратуры системы измерения, отработки параметров нейтронного потока и формирование сигналов в систему управления и защиты реактора;

- недостаточная степень резервирования;

- отсутствует автоматический контроль исправности измерительных каналов;

- в существующей системе контроля нейтронного потока переключение диапазонов изменения нейтронного потока выполняются вручную.

С целью устранения указанных недостатков предусматривается замена существующей системы на современный аппаратурный комплекс типа АКНП-7.

Комплекс АКНП-7 обеспечивает контроль нейтронной мощности и скорости ее изменения во всех режимах, формирует сигналы превышения заданных значений мощности и периода, выдает сигналы в систему управления и защиты, осуществляет регистрацию, отработку и представление информации оператору.

8. Оптимизация существующих сигналов аварийной защиты и введение дополнительных сигналов аварийной защиты реактора, в том числе:

- разделение сигналов на независимые по низкому уровню в компенсаторе объема и по давлению над активной зоне реактора.

Это позволит более однозначно обеспечить аварийную остановку реактора по потере теплоносителя первого контура;

- корректировка сигналов на включение систем безопасности. Сигналы на срабатывание систем безопасности должны быть по значению, по крайней мере, одинаковы с сигналами аварийной остановки реактора или останов реактора должен быть опережающим. Это устраняет возможность непрохождения сигнала на аварийный останов реактора из-за действия системы аварийной подпитки первого контура и спринклерной системы в боксе ПГ-ГЦН;

- введение аварийной защиты реактора по низкому уровню в парогенераторах. Это позволит отключить реактор на ранней стадии и поддержать большой запас воды при потере питательной воды, что повысит надежность отвода остаточного тепла от активной зоны;

- введение аварийной защиты реактора по высокому давлению в первом контуре. Это устраняет возможность открытия предохранительных клапанов компенсатора объема;

- введение аварийной защиты реактора по высокому уровню в компенсаторе объема. Это позволит избежать истечения водяной фазы через предохранительные клапаны КО.

9. Модернизация системы аварийной подпитки первого контура.

Необходимо выполнить не менее двух независимых каналов с резервированием активных элементов, обеспечивающих длительное охлаждение активной зоны САОЗ. Мощность насосной группы система должна обеспечить отвод тепла от активной зоны при аварии разрыва трубопровода первого контура сечением до Ду200 с односторонним истечением теплоносителя (запроектная авария).

При этом повреждение активной зоны не должно превышать максимального проектного предела повреждения твэлов в соответствии с требованиями ПБЯ РУ АС-89.

Должно быть два независимых канала САОЗ высокого давления и два независимых канала САОЗ низкого давления.

Для аварийной подпитки первого контура в аварийном режиме течи теплоносителя с наложением обесточивания должно быть установлено не менее двух гидроемкостей с рабочим давлением 55-60 кгс/см^2 и объемом борного раствора 40 м^3 в каждой гидроемкости. Гидроемкости присоединяются к разным петлям первого контура.

До реализации этого мероприятия необходимо выполнить следующие компенсирующие мероприятия:

- технические меры по предотвращению аварии с течью теплоносителя первого контура более, чем через сечение 3х26 мм (например при разрыве коллектора продувки или возврата продувки);

- повышение надежности активных компонентов существующей системы аварийной подпитки;

- обеспечить контроль появления воды в помешении насосов аварийной подпитки;

- обеспечить в режиме обесточивания запуск двух насосов аварийной подпитки в каждой из двух групп насосов за счет изменения схемы ступенчатого пуска механизмов ответственных потребителей;

- выполнить замену насосов аварийной подпитки ЭП-50 на насосы ЦН65-130.

10. Ввести систему длительного расхолаживания реакторной установки и отвода тепла от активной зоны через второй контур в соответствии с современными требованиями при всех переходных и аварийных режимах, несвязанных с течами первого контура.

Система должна иметь не менее двух независимых каналов.

До реализации этого мероприятия для снижения вероятности отказов общей причиной в дополнение к существующей системе аварийной питательной воды парогенераторов ввести дополнительную независимую систему аварийной питательной воды, состоящую из 2-х каналов. Каждый канал обеспечивает подачу питательной воды в три парогенератора. Трассировка трубопроводов должна осуществляться вне площадки на отметке 14,7 м машзала.

Реализация этого мероприятия позволит повысить надежность отвода тепла от активной зоны.

11. Внедрение системы аварийного дренирования гидрозатворов из горячих ниток трубопроводов Ду500 и аварийного газоудаления из-под крышки реактора и коллекторов парогенераторов для обеспечения циркуляции теплоносителя через активную зону при авариях с разуплотнением первого контура.

Аварийное дренирование гидрозатворов из петель Ду500 выполняется для повышения эффективности охлаждения активной зоны при авариях с разуплотнением первого контура.

12. Для исключения быстрого расхолаживания первого контура при авариях с разрывом паропроводов и для локализации аварии установить на перемычках Ду400 между паропроводами от парогенераторов и главным паровым коллектором быстродействующие запорно-отсечные клапаны (БЗОК). Два БЗОКа устанавливаются также на главном паровом коллекторе.

Система блокировок по управлению БЗОКами обеспечивает аварийную остановку реактора и автоматическое отключение дефектного участка паропроводов от остальной части второго контура и исключает быстрое расхолаживание первого контура.

13. Выполнить мероприятия по повышению надежности и эффективности спринклерной системы для снижения давления в боксах ПГ-ГЦН при аварии с течью теплоносителя первого контура через сечение Ду100 мм. При запроектных авариях должны быть предусмотрены устройства для предотвращения образования концентраций водорода в герметичных помещениях.

14. Выполнить мероприятия по повышению герметичности боксов ПГ-ГЦН и уменьшения выхода радиоактивных продуктов в окружающую среду.

15. Выполнить мероприятия по локализации и снижению аварии с течью из первого контура во второй через сечение до Ду100 мм за счет введения технических средств по управлению этой аварией.

16. Должны быть приняты меры по обеспечению целостности герметичных помещений в аварии разрыва главного циркуляционного трубопровода с истечением через 2F Ду 500 мм путем регулируемого сброса парогазовой среды через систему струйных конденсаторов.

17. Модернизация системы надежного электроснабжения с введением не менее двух независимых каналов по числу каналов систем безопасности в технологической части.

18. Создание резервного щита управления (РЩУ).

Внедрить РЩУ с выполнением минимально-необходимых функций при поражении БЩУ: аварийная остановка реактора, контроль за состоянием реакторной установки, расхолаживания энергоблока.

19. Модернизация системы противопожарной защиты энергоблоков.

Должен быть выполнен комплекс мероприятий с целью повышения пожарной безопасности с введением как пассивных мер предупреждения пожаров, так и реконструкцией активных средств пожаротушения.

20. Модернизация системы технического водоснабжения ответственных потребителей с введением двух независимых каналов с физическим разделением каналов.

До реализации этого предложения в комплексе для каждой АЭС с учетом особенностей станции должны быть реализованы первоочередные мероприятия по повышению надежности охлаждения ответственных потребителей (дизельгенераторы и др.).

21. Модернизация системы внутреннего и внешнего радиационного контроля с внедрением автоматической системы контроля радиационной обстановки на АЭС (АСКРО).

Система должна оперативно определять и прогнозировать радиационную обстановку на территории промплощадки и в районе размещения АЭС при аварии.

В настоящее время в рамках программы технической помощи со стороны ЕЭС по проекту ВВЭР-440 (В-230) Российскими институтами совместно с западными специалистами проводятся расчеты аварийных режимов по расширенному спектру аварий, а также выполняется вероятностный анализ безопасности (проекты 1.3 и 1.4 TACIS-91).

Эти анализы позволят:

- проверить правильность запланированных мероприятий;
- уточнить приоритеты в графике реализации;
- идентифицировать "слабые места" в проекте и выдать рекомендации по дополнительным мероприятиям.

BG9600403

SAFETY REASSESSMENT OF THE PAKS NPP

(The AGNES Project)

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ABSTRACT

In Hungary a nuclear power plant (Paks NPP), consisting of four units, equipped with 440MW VVER-440/V-213 type reactors is operated. The units were put into operation in the eighties. The operating experience is fairly good, no serious safety related problem occurred. However, the reassessment of the NPP's safety, according to the internationally recognized criteria of the nineties seems to be useful. That is the reason why the AGNES (Advanced General and New Evaluation of Safety) project was started, aiming the reassessment of the safety of the Paks Nuclear Power Plant.

The main objective of the AGNES Project for the reassessment of the safety of the Paks Nuclear Power Plant is to improve the safety culture of our nuclear technology. To ensure this the objectives are to be reached as follows.

- A report on the reassessment of the safety of the plant has to be prepared, by using internationally acknowledged up-to-date techniques on the level of the nineties.

- The project should include the updating of design basis accident analyses, the performing of severe accident analyses and the preparation of a level 1 probabilistic safety analysis study.

- The project should help in determining the priorities for safety enhancement and back-fitting measures and in identifying strategies for severe accident management.

- One of the objectives of the project should be the facilitating the preparation of a revised Safety Analysis Report, satisfying the requirements of the expected new Hungarian regulations.

The project has to be finished by publishing in 1994 a Final Report.

INTRODUCTION

The safety of VVER type reactors has become an important issue both at national and international level. Although serious criticism concerns only the V-230 design, it was deemed advisable to reassess the safety of the Paks NPP's V-213 units. This is partly due to certain criticism levelled at the safety of this type of unit and also to the accumulated knowledge on safety issues and safety-relevant parameters of the Paks NPP. The safety enhancement and backfitting measures already performed and planned should be critically evaluated and new measures can be proposed. In addition to this the deficiencies in the presentation and documentation of certain safety aspects should also be overcome. The aims of the work were in full agreement with the IAEA recommendation on the necessity for periodic reassessment of the safety of NPPs.

To meet these goals the Hungarian Atomic Energy Committee decided to launch the AGNES project. The project started in the autumn of 1991 and due to be completed by the end of this year. The preliminary conclusions have been summarized in 1993.

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The project has to be finished by publishing a Final Report by the end of 1994.

The Final Report of the project will be describe the following topics: licensing of the Paks NPP, site description, basic design principles, description of the safety related systems, operating and safety instructions, operational experience, approved measures for safety enhancement, system analysis and description, analysis of design basis accidents, severe accident analysis, level 1 probabilistic safety analysis. Based on the assessments some recommended measures for safety enhancement will also be formulated in the final version of the Final Report.

ANALYSES

The performed analyses can be divided into four groups

- system analysis and description,
- analysis of design basis accidents,
- severe accident analysis,
- level 1 probabilistic safety analysis.

1 System analysis and description

In the design practice of nuclear power plants specific design principles have been elaborated, to ensure that the systems needed for the safety functions should fulfil their respective functions with maximum reliability. If a plant is already in operation, the task of system-technical analyses can be formulated, as for investigating, to what extent the systems of the plant can fulfil the safety functions formulated above. This question was raised and answered by the system-technical analyses performed in the AGNES project.

Since even good quality components and equipment may cease to function in unpredictable time and manner, the single failure criterion was introduced. It is a deterministic criterion, which defines a simple design viewpoint with the aim of providing the systems with certain minimal redundancy and independence between the systems and equipment groups. The fulfilment of the single failure criterion in the Paks NPP was investigated in the project.

Results of the single failure analysis did not indicate dramatic or unknown deficiencies, i.e. the safety systems are protected against single failure. The shortcomings should be supplemented by safety enhancement measures in certain cases whereas further studies are needed in other cases.

In order to prevent the occurrence of multiple failures the diversity is applied as a specific design principle, i.e. the fulfilment of safety functions is ensured by (otherwise redundant) component groups which differ from each other in any respect. The fulfilment or lack of the diversity principle was also investigated in the project.

It was pointed out in the analysis of common mode failures that almost all analyzed component groups are well protected. Correspondingly, the components may not fail simultaneously because of a common cause. Independence of the components increases plant safety, thereby affecting PSA results. ECCS HPIS pumps form the only group whose elements are not satisfactorily protected.

As a consequence of fires, internal flooding and high energy line breaks several safety systems may fail, representing especially important common cause failures, since the fulfilment of one or more safety functions may occur to be problematic. These problems were studied in detail in the AGNES project.

Judging from the basic conclusions of fire safety evaluation the fire sections are well equipped with fire detectors but, lacking the plant specific detector sensitivity data, it is hard to assess the quality of fire detection. The studies pointed out deficiencies of specific safety design principles in some rooms.

In the rooms studied from the point of view of internal flooding, the flooding do not lead to events which would affect the availability of the investigated systems. In case of redundant systems it means that the systems are physically separated: flooding of one system does not influence the availability of the others.

Judging from the high energy line break analyses performed, certain line breaks may lead to the failure of subsystems belonging to safety systems as a dependent failure. Analysis of the HELB scenarios has not been completed but the important sensitive locations were conclusively identified.

Recently a significant effort has been devoted to the investigation of the possibility of inadvertent boron dilution and especially a diluted boron slug formation in the primary coolant of VVER-type and other pressurized water reactors. According to our study for the Paks NPP, the formation of a diluted boron slug cannot be fully excluded as a consequence of connecting an inactive loop. Though the events may be initiated by serious violation of administrative-technical rules, their further study seems to be necessary.

The containment of the VVER-440/V213 units is a special structure deviating significantly from the Western containment types. It is an important problem whether the containment structures withstand the high pressure emerging during accidents. This problem is connected to the question whether the depressurization system of the containment can function properly. According to the investigations coordinated by Paks NPP Ltd. independently of the AGNES project, the concrete structures of the containment building are able to withstand the pressures occurring at primary line breaks if the depressurization system of the localization tower operates with at least 75 % efficiency. This statement takes into account the calculational reserves and is based on the Hungarian concrete standards valid during the design period of the plant.

The fact that the containment cannot be isolated absolutely because of technological reasons and boundary penetrations are needed leads to the further important system-technical problem of containment by-pass. Containment by-pass studies pointed out, on one hand, problems caused by the steam generator collector cover opening and the means of their prevention, on the other hand, the measures needed to prevent the release of radioactivity into the environment in case of a break of lines containing primary coolant and penetrating the containment.

The consequences of external events were studied independently of the AGNES project under the coordination of the Paks NPP Ltd. The main results were as follows:

- Only earthquakes and aircraft crashes had to be studied in detail from among external events since the risk of other events was found to be insignificant.

- The earthquakeproofness of Paks NPP is the key issue from the point of view of being protected against external events. The plant was not designed for earthquakes and based on our present knowledge of site seismicity, the maximum design earthquake can be higher than supposed in the actual design. This situation is exacerbated by the lack of reliable seismic input due to the scientific problems related with site seismicity. This requires special measures from the plant operator. The lack of special knowledge, experience, regulations, and also of design information is the main problem in deciding the earthquakeproofness strategy and the execution of the realization project. On the basis of the results of works conducted with the support and under the auspices of IAEA and including domestic and foreign technical capacities, earthquakeproofness that satisfies realistic expectations can be realized. By 1995, the undergoing work that is currently in progress will provide a full insight into the actual earthquakeproofness of the plant and will also result in a basic improvement in the earthquakeproofness of the units. The earthquakeproofness project has well-defined tasks with strict deadlines for 1994-1995. Revision of earthquakeproofness strategy and the determination of further tasks in the project will become necessary once the results of site seismicity investigations and component capacity determination are known.

- The aircraft crash study was based to a considerable extent on qualitative technical considerations and stated the following:
    - Paks NPP is not designed against aircraft crash.
    - the scattered location of plant buildings in a large area is favourable from the point of view of airplane crash risk, but, on the other hand,
    - the plant's vulnerability is increased by the twin-unit arrangement and the limited spatial separation of redundant systems.
    - the various buildings and parts shield each other significantly, this is extremely important from the point of view of protecting the reactor and the spent fuel storage pool.

Because of the conservative approach, the aircraft crash frequency value obtained can be accepted (though it is higher than the target event/unit/year screening value) and its further refinement is not needed. The risk calculated for the entire plant is 10^{6} event/year. The recent IAEA consultation and assessment confirmed that aircraft crashes cannot be supposed as a major source of risk for plant safety.
Finally the radiation protection was investigated in the framework of system-technical studies. The radiation doses to people living in the vicinity of Paks NPP are below the limits of detection. The maximum population dose is less than 0.1 % of the regulatory dose limit and corresponds to 0.01 % of the average natural background dose.

2 Analyses of design basis accidents

The task of the accident analyses was to assess plant safety on the basis of internationally recognized systems of requirements by using generally accepted state-of-the-art tools.

The most advanced computer codes were applied in the reactor safety studies, and in cases where a system specific validation was needed to assure quality of results or assessing their uncertainties, they were performed. The set of initiating events used covers every initiating event considered worldwide to affect plant safety and also specific cases occurring in VVER-type reactors.

Since the plant was licensed on the basis of Soviet norms adapted at that time, a special attention was devoted in the AGNES project to defining the acceptance criteria. The acceptance criteria were elaborated by taking into account the regulations in the USA, Germany, France and Finland and they correspond to the European expectations. More strict criteria are valid for the more frequent Anticipated Operational Occurrences than for the Postulated Accidents (PA) which probably do not occur during the lifetime of the units.

The most important permissible values of key parameters as given in the criteria are presented in Tab.1.

Table 1. Limiting values of analysis key parameters

Quantity	AOO case	PA case
Minimum of departure from nucleate boiling ratio ($DNBR_{min}$)	>1.33	-
Maximum cladding temperature ($T_{clad.max}$) [°C]	-	< 1273
Maximum pressure in the primary circuit (P_{Imax}) [MPa]	<1.1*13.5	<1.35*13.5
Maximum pressure in the secondary circuit (P_{IImax}) [MPa]	<1.1*5.5	<1.35*5.5
Radially averaged fuel enthalpy (H) [J/gUO_2]	< 586	< 963

Group 1 of initiating events: Increase of secondary heat removal

The initiating events belonging to this group are listed in Tab. 2.

The steam line breaks are the most important initiating events in the given group, since in the course of the transient the shut-down reactor may become critical again. The IVO In. investigated several subcases in the framework of the project. It was found that recriticality may occur only if the steam side isolation valve does not close and the main circulation pump (MCP) does not stop. The results concerning the $DNBR_{min}$ and the minimum core inlet temperature ($T_{in,min}$) show that even in case of recriticality, not only the PA criteria but even the more strict AOO criteria are fulfilled for the whole transient. Making use of this it was proven without further analyses that the acceptance criteria are fulfilled for the other transients belonging to the given group of initiating events.

Table 2. Events leading to the increase of secondary heat removal

Initiating event	Type
1. 1 Disturbance of the feedwater system which leads to the decrease of feedwater temperature	AOO
1. 2 Disturbance of the feedwater system which leads to the increase of feedwater flow	AOO
1. 3 Disturbance or failure of the system pressure control which leads to increasing steam flow	AOO
1. 4 Inadvertent opening of the steam generator relief or safety valve	AOO
1. 5 Spectrum of steam line breaks inside and outside the containment	PA

Group 2 of initiating events: Decrease of the secondary heat removal

In this group of initiating events the $DNBR_{min}$ and the maximum pressure in the primary and secondary circuits are the relevant parameters of criteria. The investigated events are given in Tab.3.

In case of feedwater line breaks not only the PA but also the more strict AOO criteria are fulfilled. During the process the primary circuit may cool down to a dangerous level leading to recriticality, if the $\Delta p > 5$ bar steam generator protection signal is not actuated.

Table 3. Events leading to the decrease of secondary heat removal

Initiating event	Type
2.1 Disturbance of the steam pressure control which leads to decreasing steam flow	AOO
2.2 Loss of electricity	AOO
2.3 Turbine trip	AOO
2.4 Inadvertent closure of the main steamline isolation valve	AOO
2.5 Loss of condenser vacuum	AOO
2.6 Simultaneous loss of off-site power and internal A.C. supply	AOO
2.7 Loss of normal feedwater flow	AOO
2.8 Feedwater line breaks	PA

Group 3 of initiating events: Decrease of primary coolant inventory

The same quantities are relevant with respect to this group of initiating events as in the previous one, with the exception of case 3.4 which is related to a refuelling state.

Table 4. Parameters of transients leading to the decrease of primary coolant inventory

Initiating event	Type
3.1 Loss of one or more MCPs	AOO
3.2 MCP seizure	PA
3.3 MCP shaft break	PA
3.4 Disturbance of natural circulation	AOO

The AOO criteria are fulfilled in case of initiating events 3.1 to 3.3. In case 3.4 which may emerge during refuelling as a consequence of the failure of the decay heat removal system, the natural circulation and the heat exchange in boiling condition ensure that the cladding temperature does not reach 400 K. The operator has one hour time for an intervention to restart natural circulation without violating the criteria.

Group 4 of initiating events: Reactivity and power distribution anomalies

The initiating events belonging to this group are presented in Tab.5.

Consequences of initiating events 4.3, 4.5 and 4.6 were studied by reactor static and thermohydraulical calculations which showed that the criteria are fulfilled. For the other cases detailed coupled kinetic calculations were made (partially by IVO In.). The relevant parameters for AOO cases are the radially averaged enthalpy (H), the $DNBR_{min}$ and the maximum pressure in the primary circuit (P_{Imax}). The results show that the criteria are fulfilled both in the AOO and PA cases, but in case 4.7 a permitted fuel cladding failure occurs to a significant but small extent.

Table 5. Reactivity induced accidents and anomalies of power distribution

Initiating event	Type
4.1 Uncontrolled withdrawal of a control assembly in a subcritical or low-power start-up state	AOO
4.2 Uncontrolled withdrawal of a control assembly group at different power levels	AOO
4.3 Malfunction of a control assembly	AOO
4.4 Connection of an inactive loop by mistake	AOO
4.5 Maloperation of the volume and boron control systems	AOO
4.6 Loading a fuel assembly into a wrong position	PA
4.7 Spectrum of control assembly ejections	PA

Group 5 of initiating events: Increase of reactor coolant inventory

The essential parameters in this group are the same as in groups 2 and 3. It can be stated that the AOO criteria are fulfilled.

Table 6. Events leading to the increase of reactor coolant inventory

Initiating event	Type
5.1 Inadvertent ECCS operation at power	AOO
5.2 Maloperation of the volume and boron control systems which leads to the increase of reactor coolant inventory	AOO

Group 6 of initiating events: Decrease of reactor coolant inventory

The initiating events belonging to this group are listed in Tab.7. Numerous subcases were studied (partially at GRS). The relevant thermohydraulical parameter in this type of events is the cladding temperature.

In cases 6.3 and 6.4 the cladding temperature decreases as a result of the initiating event. While events 6.2 and 6.6 do not have any thermohydraulical consequences.

Table 7. Accidents leading to the decrease of reactor coolant inventory

Initiating event	Type
6.1 Inadvertent opening of pressurizer safety or relief valve	PA
6.2 Primary circuit line break outside the containment	AOO
6.3 Steam generator tube break	PA
6.4 Steam generator collector cover opening	PA
6.5 Spectrum of loss of coolant accidents	PA
6.6 MCP leak into the intermediate circuit at loss of power	PA

In order to assess the severity of loss of coolant accidents (including loss of coolant during control assembly ejection) containment behaviour studies and dose calculations were performed.

The DBAs investigated with respect to these two problematic are as follows:
- inadvertent opening of the pressurizer safety valve,
- spectrum of loss of coolant accidents,
- steam line break,
- control assembly ejection,
- steam generator collector cover opening.

The investigated cases can be summed up as follows:

In the case of the 200 % cold-leg break the resulting maximum pressure was smaller than the design value. From the point of view of containment stresses streamline break has proved to be more serious than the 200 % LOCA, but the calculated value of maximum pressure does not exceed the limits even in this case.

The relevant criteria are satisfied in events having radiological consequences, as was expected. The environmental doses were calculated using models and methods fully complying with the CEC recommendations. For the case of steam generator collector cover opening one could assume that serious problems may occur because of the direct environmental release, but the analyses pointed out that release of the primary coolant activity does not lead to the environmental dose limits being exceeded.

According to calculations, not even the most threatened group of the population living in the surroundings is exposed to doses exceeding the established limits.

Hydrogen generation was not taken into account in the plant design. The eventual generation of gases in the vessel makes the establishment of a gas removal system in the primary circuit necessary, but it is even more important to establish a system for hydrogen removal from the containment.

The actual results of the DBA analyses verified the expectations concerning the large safety margins of the VVER-440/V213 system. For the majority of the initiating events considered, the plant satisfies the criteria even for the minimum configuration of safety systems.

Group 7 of initiating events: ATWS analyses

The analyses of transients with the failure of reactor scram play a special role. The analysis of these ATWS cases has been only recently included in Safety Analysis Reports and thus such analyses have been scarcely made for VVER type reactors. In these analyses the fulfilment of PA criteria was required and instead of the usual conservative assumptions on input data the nominal (best-estimate) values were used. It was a very important result of the analyses that in course of ATWS events no dangerous state of the system occurs and the PA criteria are satisfied.

The investigated ATWS events are listed in the Tab.8.

Table 8. Analyzed ATWS cases

Case	Type
Inadvertent withdrawal of a control assembly group	PA
Loss of feedwater	PA
Loss of off-site power	PA
Trip of both turbines	PA
Inadvertent closure of the main steam isolation valves	PA

Group 8 of initiating events: Pressurized thermal shock (PTS) analyses

The PTS transients require also special investigations since fracture mechanical calculations have to be performed after the thermohydraulic analysis in order to verify whether the acceptance criteria are fulfilled for the actual pressure vessel material. The vessel material properties are enworsed with time as a consequence of radiation damage therefore the fulfilment of the acceptance criteria should be proven for the material data extrapolated to the end of planned lifetime.

After 24 years of operation the safety factors against crack initiation are greater than unity, but according to the conservative calculations the 40 years lifetime seems to be problematic for every unit. The problems can be handled by appropriate lifetime management. Recommendations were elaborated in the project concerning this issue.

The initiating events studied for every unit were the following ones.

Table 9. Analyzed PTS cases

Case
Inadvertent opening of the pressurizer safety valve
Different medium size LOCAs
Opening of steam generator collector cover
Large break LOCA
Steam line break
Inadvertent ECCS operation

3 Severe accident analyses

Severe accident analyses of a VVER-440/V213 type unit provide an opportunity to draw conclusions in four areas: on the in-vessel phase of accidents, on the containment phase of the accidents, on the radioactivity released into the environment, and on accident management.

The time history of the in-vessel phase of severe accidents was investigated with the STCP code package and is shown in Tabl.10. for three representative cases:

- large break LOCA combined with the loss of off-site and on-site power (LBLOCA),
- small break LOCA combined with the loss of off-site and on-site power (SBLOCA),
- loss of off-site and on-site electric power (LOEP).

Table 10. Important events during the in-vessel phase of severe accidents

Event	LBLOCA (min)	SBLOCA (min)	LOEP (min)
Core uncover	15	181	491
Beginning of fission product release from fuel	26	208	513
Melting starts	30	212	520
Beginning of core slump	44	224	551
Beginning of gridplate heat-up	46	227	582
End of core slump	44	224	551
Gridplate fails	73	250	599

As far as the in-vessel phase is concerned, the VVER-440 reactor has extremely large water reserves both on the primary and the secondary side. If these reserves are expressed in terms of time, it means that more time passes until the core melts than in the case of reactors of other designs. For example, for an accident scenario initiated by station blackout this time is 6 hours, which permits more flexibility for the possible interventions. The relatively small reactor core, where the power density is low, is situated in a long reactor vessel, which contains the control assembly follower fuel elements with a lot of iron structures below the core. The six-loop arrangement leads to a high volume primary circuit and this ensures that the core or the pressure vessel remains intact for a longer period even if the core remains uncooled.

If the core melting cannot be prevented then the molten corium attacks the material of the reactor vessel. It was found that the vessel damage probability of a VVER type reactor vessel is lower than or equal with the probability for some other types.

In the course of severe accidents the basic barriers isolating the accumulated activity from the environment (fuel matrix, fuel cladding, primary circuit with the reactor vessel) may damage and the containment provides the last barrier from the point of view of radioactive releases to the environment. A significant amount of activity may be released into the environment via the normal leakage of the containment but the situation is even more dangerous in the case of the physical damage of the containment. The analyses showed that the pressure load to the containment in the in-vessel phase of severe accidents does not exceed the design value even if the localisation system is available only up to a limited extent (75 %). Since our knowledge on the failure pressure of the containment is not satisfactory, the investigations should be continued.

As far as the containment phase is concerned, the following endangering mechanisms should be mentioned for the VVER-440/V213.

- If vessel failure occurs at high primary system pressure, then the high pressure in the shaft may lead to the loss of containment integrity. Therefore the pressure reduction of the primary circuit is of primary importance in case of high pressure core melting.
- In the course of severe accident sequences, the dominant hydrogen sources are zirconium oxidation (in the in-vessel phase) and core-concrete interaction (in the containment phase). The danger of hydrogen burns depends on the hydrogen concentration. Due to high hydrogen concentration occurring in certain sequences, detonations (shock waves) may occur leading to high pressures. At lower concentrations turbulent burns may develop with intensities depending on the actual

burning mechanism. From among them higher pressure occur in adiabatic full burning and in case of so called accelerated flames spreading from one room to another.

With low pressure core melting, damage to the reactor shaft due to molten corium is the source of essential danger. The long-term pressure transient initiated by the core-concrete interaction is not a basic risk for units with high containment leakage rate, but in case of low leakage rate this may be problematic.

Assuming releases only via containment leakage even for severe accidents, calculations were made for the retention of the reactor hall, the activities released into the environment and their dose consequences. According to the calculations no fatalities occur due to deterministic injuries, the threshold doses are not exceeded for any human organs. The release of 137Cs does not exceed 100 TBq within the error margin of the calculation and the late consequences of other radionuclides remain below the effective dose equivalent caused by 137Cs release. It means that the Finnish requirements are fulfilled for severe accidents. At the same time it should be made clear that this statement relates to cases when the containment integrity is not damaged as a result of a severe accident. This circumstance provides a justification for continuing the containment failure studies.

The main objectives of accident management are to prevent the effects of beyond design basis accidents and to mitigate their consequences. Safety objectives can be formulated, and if they are achieved the severe accident sequence can be prevented or at least its consequences can be mitigated. These objectives are as follows:

- prevention of core damage,
- prevention of reactor vessel failure,
- maintaining containment integrity,
- reduction of radioactive releases into the environment.

Accident management is based on the application of the following strategies associated with safety objectives:

- prevention of core melting by operator interventions,
- primary bleed in case of high pressure core melt sequences,
- prevention of containment failure as a consequence of hydrogen burns,
- prevention of reactor shaft failure,
- prevention of containment overpressurization,
- minimization of radioactive releases from the containment.

The order above corresponds to the priority of the above strategies.

4 Probabilistic safety analyses

The objectives of the current probabilistic safety assessment (PSA) were restricted to Level-1 PSA which

- quantifies the core damage frequency,
- identifies and quantifies the most significant event sequences leading to core damage,
- identifies and quantifies the most important influencing factors of safety,
- estimates the uncertainty level of the results.

The scope of the PSA investigations included internal initiating events, no external and internal hazards were treated. It was related to the full power operation of the plant, and only the reactor core was considered as potential source of radioactive release.

Approach and methods applied

Generally the procedural framework given in the basic IAEA (50-P4, 1994) and NRC (NUREG/CR-4550, 1990) documents was considered and followed. For probabilistic quantification of events and event sequences the RISK SPECTRUM PSA Code Package (RELCON Teknik AB, Sweden, 1992) was used. A multi-step task-list was defined for the PSA part of the Project, the approach and methods applied for each task are listed as follows.

a. Initiating event identification

A broad-scope generic list was modified based on plant-specific experience and simulator experiments, their grouping was finalized based on the thermohydraulic simulations. The final list includes 53 initiators. Frequencies of the initiating events were assessed based on the Bayesian integration of the generic and plant-specific data.

b. Event sequence analysis

Small event tree/large fault tree concept was applied. The event trees were set up during and Expert Panel, the transient ones were verified by simulator experiments. the LOCA ones were verified by transient simulations.

c. System analysis

The modular fault tree structure was used making the gradual integration of I&C and electrical circuit fault trees possible. Component and system boundaries, as well as failure modes were defined a-priori the analysis and applied by each working group.

d. Dependent failure analysis

The common cause failures of components of the same type were considered with beta-values. The beta-parameters used were calculated as ratio of generic common cause failure rates and plant specific total ones for the technological components, for I&C and electrical components generic values were adapted.

e. Human reliability analysis

The modelled human failures are categorized as failed interactions before and after the initiating event. For the first category the ASEP HRA procedure was applied. the HIs of the second category were quantified based on operator reliability experiments on the full-scope simulator of the plant assuming HCR models and developed operator decision trees.

f. Reliability data base development

For technological components an integration of the generic and available plant-specific data using Bayesian updating was performed, for I&C and electrical components mainly plantspecific generic reliability data were included.

g. Accident sequence quantification

The point estimation of the core damage and event sequence frequencies was done by the RISK SPECTRUM PSA Code Package. Effects of differences in the technological components and design of the four plant units were also assessed.

h. Sensitivity and uncertainty analyses

They were performed by the RISK SPECTRUM PSA Code. The sensitivity studies were concentrated on specific safety-oriented issues. During the uncertainty analysis mainly lognormal distributions of the input reliability parameters were assumed.

Results of accident sequence quantification

According to the PSA results the core damage frequency in the present state of the unit exceeds the target value for new reactors under design or construction, but the magnitude is in the range of those values known for operating plants commissioned several years ago.

Two initiating event group dominate the situation. The feedwater and steam line breaks leading to the total loss of feedwater are the main contributors to the core melt frequency. According to this the most important core melt frequency contributor initiating events are the feedwater collector rupture and the main steam collector rupture.

In case of full separation of the Auxiliary Feedwater System and/or the introduction of primary feed and bleed procedures would significantly decrease the present core damage frequency under the present modelling conditions. The effect of this two safety enhancement measures manifests itself also in the more equalized contribution of initiating events to the core damage frequency, whenever they are introduced.

From among the factors determining the core damage, the effect of eventual human failures is the most significant, while the contribution of other failure types (such as common mode failures, I&C and electric supply failures) to the overall risk would relatively increase if the safety enhancement measures considered in the calculations were to be accomplished. After separating the Auxiliary Feedwater System the contribution of human failures or erroneous human interventions would reduce about 10%.

Because of the importance of the human factor, the following items are of primary importance in increasing the safety level of the plant:

- given that the maintenance procedures are trained, the probability of remanent errors should remain at the present favourable level or should even decrease.
- the operational staff who manage accidents should be trained systematically and the accident situations should be selected for inclusion into the training program by concentrating on the event sequences dominating the risk and their prevention,
- the Emergency Operating Procedures should promote effectively the management of complicated accident scenarios (e.g. by introducing a system of symptom-oriented procedures) and they should cover as widely as possible the prevention of sequences leading to core melting (e.g. by introducing accident management procedures).

In case of separating the Auxiliary Feedwater System the failures of the I&C system have a relatively significant contribution to the core damage frequency. During the planned reconstruction of this system, it should be set as an objective that the man-machine interface and the hardware and software tools of safety related I&C support effectively the quick recognition of accident situations and the prevention of severe consequences on the basis of both automatic and human intervention. The component and system functions relevant from the point of view of risk reduction and the priorization of the planned reconstruction can be determined on the basis of the PSA models elaborated in the framework of the AGNES project.

The contribution of high pressure sequences was investigated separately since these sequences represent a major danger to containment integrity according to the severe accident analyses and such scenarios may lead to catastrophic radioactive releases in case of containment failure. The safety enhancement measures proposed by the PSA studies would reduce not only the core melt
frequency, but the proportion of high pressure sequences as well.

RESULTS

The results, summarized in this paper originate both from the draft version of the Final Report assessing the safety of the Paks NPP and from the Executive Summary of the AGNES Project.

The Final Report is a summarizing analysis that contains sufficient information for judging the nuclear safety of Paks Nuclear Power Plant, together with its background information, the document enables an accurate judgement to be made of the nuclear safety of the power plant.

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ОПЫТ НАУЧНОГО РУКОВОДСТВА ПУСКОМ И СОПРОВОЖДЕНИЯ ПРИ ЭКСПЛУАТАЦИИ ЭНЕРГОБЛОКОВ АЭС С ВВЭР

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ВНИИАЭС, как ведущий институт по научно-техническим вопросам ввода в эксплуатацию и эксплуатации АЭС накопил большой опыт работ по совершенствованию пусковых и эксплуатационных технологических процессов на АЭС, повышения их безопасности и надежности.

Специалисты ВНИИАЭС выполняли работы по научному руководству пусками на большинстве энергоблоков АЭС в России, Украине, Болгарии, Венгрии и Чехословакии. Выполнено большое число разработок по модернизации проектов. В основе анализа и принятия технических решений использован комплексный подход к работе оборудования и систем энергоблока в целом.

Идея комплексного подхода пронизывает все программы пусконаладочных работ, программы модернизации энергоблоков и программы совершенствования процедур эксплуатации. При определении содержания вышеназванных работ применен принцип оптимизации эффекта от затрат, для чего использованы экспертные оценки. Приведенены конкретные примеры описанного подхода.

1. Пуско-наладочные работы на АЭС.

Пуско-наладочные работы требуют системного подхода на всем их протяжении. Благодаря этому удается оптимизировать график подготовки и проведения работ на этапах пуска и получить максимум информации от испытаний.

Поэтапная программа пуска и освоения мощности энергоблока, например [1], составлена с применением системного подхода. Документальная готовность и готовность персонала к пуску учитываются наряду с технической готовностью, программа одного этапа учитывает задачи последующих этапов.

Программа физического пуска помимо решения чисто физических задач (вывода реактора в критическое состояние, измерение нейтронно-физических характеристик) включает значительный объем работ по наладке КИП и А, СВРК, по проверке эффективности систем расхолаживания, системы аварийного ввода бора, по измерению гидравлических характеристик первого контура, комплексному опробованию СУЗ и др., т.е. то что потребуется для последующих этапов пуска. Еще более широкий круг взаимосвязанных задач решается на этапах энергопуска: тарировка АКНП, наладка и испытание автоматических регуляторов энергоблока, проверка работы информационных систем и др. Особая роль отводится динамическим испытаниям (сброс - наброс нагрузки, отключение оборудования). Именно они позволяют проверить работу оборудования в различных режимах и энергоблока в целом и оценить качество совместной работы многих систем энергоблока.

Следует отметить, что пуск энергоблока, как правило, не удается выполнить строго по заранее составленным программам. Опыт показывает, что на практике необходимо учитывать целый ряд обьективных и субъективных факторов.

Неполная готовность систем и оборудования к соответствующему этапу пуска, изменение условий в энергосистеме, отказы систем и оборудования приводят к необходимости рассматривать и решать вопросы о частичном изменении программ испытаний, изменении последовательности их выполнения, изменении критериев успешности испытаний и т.п.

Ясно, что в этой ситуации роль системного подхода особенно важна. Путем экспертных оценок, проведения дополнительных расчетов, сопоставления результатов, допускается или отвергается возможность отклонений от программы. В случае признания допустимости отклонений разрабатываются, как правило, компенсирующие мероприятия (в особенности в части, касающейся безопасности). принимается согласованное в установленном порядке техническое решение или даются письменные указания за подписью представителей Главного конструктора, научного руководителя пуска, главного технолога и руководства АЭС.

Таким образом, например, была откорректирована поэтапная программа пуска энергоблока N 6 АЭС Козлодуй по результатам динамических испытаний со сбросом электрической и паровой нагрузки при освоении номинальной мощности, корректировались таблицы допустимых режимов эксплуатации РУ, были определены временные компенсирующие мероприятия при работе энергоблока с теплообменниками САОЗ, имеющими пониженную эффективность.

В ряде случаев для обоснования проводимых корректировок выполняются научные исследования и разработки. Например, на АЭС Пакш при пуске энергоблоков N 3 и N 4 с целью уточнения тепловой мощности реакторов и подготовки корректной таблицы допустимых режимов эксплуатации реакторной установки была разработана совместно с персоналом АЭС методика экспериментальной оценки радиационного нагрева термопар СВРК и байпасной протечки теплоносителя в реакторе [2]. При пуске энергоблока N 6 АЭС Козлодуй были проведены исследования по выявлению причин:

- повышенных пульсаций перепада давления на реакторе,
- пониженной эффективности теплообменников аварийного расхолаживания (для обоснования их вскрытия и очистки),
- повышенной концентрации лития в теплоносителе I контура и другие.

Для информационного обеспечения таких исследований во ВНИИАЭС созданы банки данных по нейтронно-физическим характеристикам ВВЭР и по переходным процессам. Данные имеются как для двухгодичных так и трехгодичных топливных загрузок. Для расчетного моделирования работы активной зоны и энергоблока в целом имеются необходимые коды (БИПР, АЛЬБОМ, ДИНАМИКА, RELAP).

2. Сопровождение при эксплуатации АЭС.

Задачу сопровождения при эксплуатации энергоблоков можно коротко сформулировать следующим образом: сбор информации о работе систем и оборудования энергоблоков и выдача предложений по их совершенствованию, а также совершенствованию процедур эксплуатации, в том числе энергоблока в целом.

При системном подходе к решению этой задачи должны учитываться требования нормативно-технической документации, программы технической политики в отрасли, экономический и человеческий факторы.

Систематический сбор и обработка информации о работе систем и оборудования энергоблоков выполняется ВНИИАЭС с участием эксплуатируемых АЭС России и Украины. ВНИИАЭС располагает наиболее полной базой данных о нарушениях в работе оборудования и показателях работы АЭС. С учетом этих данных подготовлены программные документы по модернизации оборудования АЭС.

Программы модернизации энергоблоков, выполняемые с целью повышения их безопасности и готовности [3] сформированы с применением системного подхода. Каждое предложение по модернизации проанализировано на безопасность не только прямого эффекта, ради которого оно внедряется, но и на безопасность побочных эффектов. Эти анализы выполнены как экспертным путем, так и с привлечением детерминированных, а в ряде случаев и вероятностных оценок безопасности.

Примером такого подхода может быть обоснование режима ускоренной разгрузки блока (УРБ). Для доказательства эффективности и безопасности этого режима в рамках детерминированного подхода выполнена серия нейтронно-физических, теплогидравлических, динамических и вероятностных расчетов [4]; проведены испытания на энергоблоках Южно-Украинской, Калининской, Ровенской и других АЭС. Эта работа была выполнена для энергоблоков с двух и трехгодичными топливными загрузками.

Другими примерами системного подхода могут быть расчетно-экспериментальные обоснования повышения динамической устойчивости энергоблоков в режимах с нарушениями расхода питательной воды в парогенераторы [5] и в режимах с непреднамеренными ("ложным") закрытием БЗОК. [6].

Приведенные три работы являются лишь частью большой программы анализа и оптимизации технологических защит и блокировок реакторного и турбинного отделений с целью повышения динамической устойчивости энергоблока в целом.

Опыт эксплуатации энергоблоков показывает, что важно обнаружить симптомы и заранее предупредить развитие аварии, что позволит снизить негативные последствия.

Поэтому во ВНИИАЭС значительное влияние уделяется внедрению систем оперативной диагностики на действующих энергоблоках (главным образом с реактором ВВЭР-1000). В основу развития систем оперативной диагностики действующих энергоблоков заложен принцип открытости для дальнейших совершенствований. Оснащенное локальными системами оперативной диагностики составляет первый этап работ. На втором этапе планируется разработать методики обработки совокупной диагностической информации с целью более достоверного выявления конкретных аномалий работы энергоблока. При этом информация от локальных систем оперативной диагностики и информация об основных технологических параметрах энергоблока поступает в систему поддержки оператора, где будет осуществляться выработка критериев для принятия решений оперативным персоналом. Работа выполняется совместно с Главным конструктором РУ и фирмой Сименс.

Выше были перечислены основные направления работ по модернизации энергоблоков с ВВЭР, выполняемые ВНИИАЭС. Кроме этого ведутся работы по повышению помехозащищенности оборудования СУЗ, по внедрению информационно-сервисной системы контроля и диагностики блоков УКТС при ремонте и др.

Процедуры эксплуатации энергоблоков совершенствуются путем корректировки технологических регламентов безопасной эксплуатации, с учетом обновляемых требований нормативно-технической документации и модернизации проекта. Эта работа выполняется совместно с организациями Главного конструктора РУ и Научного руководителя проекта.

Повышение безопасности и эффективности работы АЭС ведется также за счет совершенствования технического обслуживания (ТО) и ремонта, которое включает в себя планирование, подготовка и осуществление ТО и ремонта, послеремонтный анализ, управление материальными ресурсами, необходимыми для выполнения ТО и ремонта. При этом совершенствуется документация (организационная, нормативная, технологическая), информационная база (перечни оборудования), запасных частей, технологических средств, нормативов, сведения о дефектах), программное обеспечение, способное решать задачи планирования ТО и ремонта и управления запасными частями и материалами. В настоящее время разрабатывается интегрированная программная оболочка, которая может быть привязана к локальной сети ПЭВМ АЭС.

Сбор и анализ информации о работе и состоянии оперативного персонала, как элемента в цепях управления энергоблоком, также предполагается при системном подходе. Поэтому важными задачами эксплуатационного сопровождения нами рассматриваются следующие:

- подготовка оперативного персонала, в том числе обучение на тренажерах различного уровня,

- оценка профессионального и психофизического состояния оперативного персонала и разработка программ его реабилитации.

Эти задачи в последнее время привлекают к себе все большее внимание. ВНИИАЭС планирует такие разработки, и часть из них находится в стадии решения.

3. Опыт расчетного анализа безопасности.

Работы, выполняемые ВНИИАЭС по анализу и разработке предложений по совершенствованию эксплуатационных режимов АЭС Козлодуй рассматриваются как комплексные исследования.

Накопленный опыт выполнения анализа и обоснования безопасности в ходе работ по внедрению технических предложений, направленных на повышение безопасности, надежности и экономичности АЭС Козлодуй позволяет сделать некоторые общие замечания.

Первое: В основе подготовки и реализации технических предложений положен принцип выполнения анализа и обоснование безопасности на уровне не ниже чем в проектных технических материалах. Данный подход требует серьезной проработки проекта и критический анализ современных и исторически сложившихся методик и концепций безопасности.

Второе. Разработка мероприятий и обоснование безопасности выполняется с использованием нескольких независимых расчетных программ верифицированных по реальным переходным процессам.

Разработка пакетов исходных данных расчетных программ ведется планомерно с учетом накопленного опыта расчетного анализа и сопоставления различных режимов. Расчетный анализ опирается на следующие программы: ДИНАМИКА-5, ТЕЧЬ-М - как основа проектного обоснования безопасности; комплекс программ ВНИИАЭС для специального анализа элементов оборудования и технологических схем; программа RELAP-5, как наиболее универсальная и широко распространенная во многих странах для анализа АЭС с корпусными реакторами.

Третье. Авторский контроль за результатами внедрения на АЭС технических решений, оценка опыта эксплуатации. При необходимости корректировка и модернизация технологических алгоритмов.

Вышеназванные замечания иллюстрируются в следующих работах ВНИИАЭС, получивших внедрение на АЭС „Козлодуй" , в том числе:

- разработка системы ускоренной разгрузки блока (УРБ) для ВВЭР-1000;

- совершенствование алгоритмов работы узла водопитания парогенератора ВВЭР-1000;

- разработка технологической процедуры при ложном срабатывании БЗОК для ВВЭР-1000;

- анализ условий срабатывания разрывной защиты второго контура для ВВЭР-1000;

- разработка и обоснование режима работ ВВЭР на мощностном эффекте реактивности;

- анализ технологических процедур в режиме с полной потерей питательной воды для ВВЭР-440.

Заключение

1. Опыт пуска энергоблоков и сопровождения их эксплуатации убедительно доказывают эффективность и целесообразность системного подхода. Это относится и к процессу организации работ на энергоблоке как на этапе пуско-наладочных работ так и промышленной эксплуатации АЭС.

2. Отмечается повышенное внимание к качеству технических обоснований для тех или иных решений. Возрастает роль расчетного анализа и современных методов и концепций оценки безопасности.

3. Расширяются возможности оценки опыта эксплуатации, совершенствование технологических процесссов в ходе накопления и обобщения информации об эксплуатации. ВНИИАЭС рассматривает эти задачи как приоритетные в своей деятельности.

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ОБОСНОВАНИЕ ЭКСПЛУАТАЦИОННЫХ ПРЕДЕЛОВ ВВОДА РЕАКТИВНОСТИ ПРИ ПУСКЕ РЕАКТОРОВ ВВЭР-1000

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АННОТАЦИЯ

Рассмотрены вопросы безопасности пуска ВВЭР. Дано описание методики и программных средств по определению допустимых скоростей ввода реактивности при пусках реакторов ВВЭР-1000 в любой момент кампании и включают в себя расчет мощности источника нейтронов в активной зоне реактора в подкритическом состоянии и расчет относительной плотности нейтронов и периода реактора при достижении критического состояния.

Обсуждены вопросы регламентации и оптимизации допустимых скоростей ввода реактивности и параметров средств воздействия на реактивность при пуске реакторов ВВЭР-1000. Обосновывается возможность увеличения расходов подачи чистого конденсата в I контур при выводе реактора в критическое состояние относительно ограничений, предписываемых технологическим регламентом эксплуатации ВВЭР-1000.

Безопасный пуск реактора является сложной и ответственной задачей при вводе в эксплуатацию и в процессе эксплуатации АЭС с ВВЭР. Как известно, пуск реакторов типа ВВЭР осуществляется последовательным выполнением операций: вначале, по извлечению из активной зоны групп органов регулирования СУЗ, а затем, по снижению концентрации борной кислоты (CH_3BO_3) в теплоносителе I контура.

На ВВЭР-1000 проектом предусмотрено наличие штатной аппаратуры контроля подкритичности активной зоны остановленного реактора и в процессе его пуска. При отсутствии контроля подкритичности активной зоны безопасный пуск реактора, без нарушений условий нормальной эксплуатации, достигается прежде всего за счет ограничения скорости ввода положительной реактивности. В типовых технологических регламентах безопасной эксплуатации ВВЭР-1000 (ТРВ-1000-1 и ТРВ-1000-3) приведены следующие технологические ограничения по вводу положительной реактивности при пуске реактора:

- пошаговое извлечение групп регулирования СУЗ с выдержкой времени между шагами;
- непревышение максимально-допустимых значений расхода подпитки I контура чистым конденсатом 40 т/час, 20 т/час, 10 т/час для различных интервалов значений CH_3BO_3 в реакторе.

Данные ограничения, распространяющиеся на пуски реактора во всех состояниях, не содержат необходимых согласно требований п. 2.4.8 "Правил ядерной безопасности реакторных установок атомных станций ПБЯ РУ АС-89" указаний о величине максимально-допустимой скорости ввода реактивности и недостаточно обоснованы. В частности, ограничение расхода подачи чистого конденсата в I контур величиной 10 т/час в пусковом интервале значений CH_3BO_3 (начинается с величины CH_3BO_3 в теплоносителе I контура на 1 г/кг более ожидаемой критической CH_3BO_3) для пусков реакторов с активной зоной, работавшей на мощности, чрезмерно консервативно и приводит к значительным потерям времени на выполнение операций по водообмену, особенно в конце кампании работы реактора.

Для обоснования безопасных эксплуатационных пределов по скорости ввода реактивности и оптимизации (минимизации по времени) операций по пуску реактора из различных состояний в Нововоронежатомтехэнерго разработана "Методика определения допустимых скоростей ввода реактивности при выводе в критическое состояние реакторов ВВЭР-1000 (согласована и рекомендована к внедрению ИАЭ им. Курчатова, ОКБ "Гидропресс", ВНИИАЭС и утверждена руководством 27ГУ МАЭП).

В соответствии с методикой определяются величины эксплуатационных пределов скорости ввода положительной реактивности при выводе реакторов типа ВВЭР в критическое состояние:

- первый предел, зависящий от мощности источника нейтронов в активной зоне и величины периода реактора при достижении минимально-контролируемого уровня мощности реактора;
- второй предел, зависящий от скоростной эффективности регулирующей группы ОР СУЗ (при перемещении вниз с рабочей скоростью).

Ограничение вторым пределом связано с тем, что критическое состояние реактора типа ВВЭР достигается при снижении концентрации борной кислоты в теплоносителе первого контура, а значения первого предела могут достигать величин, близких к скоростной эффективности ОР СУЗ.

Величина первого предела скорости ввода реактивности определяется на основе требования пункта 5.5 "Правил ядерной безопасности атомных электростанций" ПБЯ 04-74, действовавших во время проектирования, сооружения и ввода в эксплуатацию большинства действующих в настоящее время энергоблоков АЭС с ВВЭР; при этом используется минимальное значение периода реактора, с которым достигается МКУ, равное 50 сек. вместо 43 (соответствующих периоду удвоения 30 сек, указанному в п. 5.5 ПБЯ 04-74.

В Методике за счет ряда консервативных приближений и построения последовательности расчета сложная задача расчета мощности источника нейтронов и определения допустимых скоростей ввода реактивности сведена к достаточно простым вычислениям с использванием данных общего применения, получаемых в комплексе нейтронно-физических расчетов (в объеме установленной номенклатуры расчетов) и при контроле за работой топливных загрузок.

Алгоритм расчетов по Методике реализован в комплексе программных средств, поставленных на персональном компьютере типа IBM PC, состоящим из:

1. Программы RNMKU, предназначенной для рсчета зависимостей относительной плотности потока нейтронов в активной зоне и периода реактора от

веденной реактивности, меняющейся во времени. Зависимость реактивности ρ (t) от времени t в программе RNMKU задается в виде:

$\rho = \rho_0 + \gamma \cdot t$, где

ρ_0 - начальная реактивность

γ - скорость ввода реактивности.

Относительная плотность нейтронов (n_k^i)в зависимости от времени ($t_k = k \cdot \tau$) вычисляется по рекуррентной формуле, полученной из уравнений кинетики /1/ с использованием кусочно-линейного приближения функции плотности нейтронов в активной зоне:

$$n_k^i = \frac{\rho_0^l - \sum_i Д_{i,k}^l - n_{k-1}^l \cdot l(\tau \cdot \beta)}{\rho_k^i - C^l - l(\tau \cdot \beta)}, \text{где}$$

$$Д_{i,k}^l = Д_{i,k-1}^l \cdot q_i + n_{k-1}^l \cdot A_i^l,\ k = 1,2,\ldots$$

$$q_i = \exp(-\lambda_i\ \tau);\quad C^l = \Sigma_i^6 C_i^l;\quad C_i^l = \beta_i(1 - q_i)/(\beta \cdot \lambda_i \cdot \tau);$$

$$A_i = C_i^l(1 - q_i);\quad Д_{i,0}^l = C_i^l;\quad n_k^l = n_k / n_0;\quad n_k^l = n_k / n_0;\quad n_k^l = n_k / n_0;$$

$$\rho_k^l = \rho_k / \beta;\qquad \beta = \sum_i^6 \beta_i;\qquad \rho_0^l = \rho_0 / \beta;$$

n_0 - равновесная (стационарная) плотность нейтронов при $\rho_0 \leq 0$;

β_i - эффективная доля запаздывающих нейтронов I-ой группы;

c_i - концентрация запаздывающих нейтронов I-ой группы;

C - сумма концентрации запаздывающих нейтронов;

l - время жизни мгновенных нейтронов;

τ - временной шаг расчета.

Период реактора вычисляется по формуле:

$$T = \frac{(n_k^i + n_{k-1}^i)/2}{n_k^i - n_{k-1}^i} \cdot \tau$$

В результате расчета по программе RNMKU определяются зависимости значений n/n_0 и T, достигаемые в критическом состоянии (при $\rho = 0$) и значений n/n_0 в момент достижения заданного значения периода от скорости ввода реактивности.

2. Комплекса программ IST, предназначенного для расчета значений мощности источника нейтронов в активной зоне остановленного реактора ВВЭР-1000, допустимой скорости ввода реактивности при выводе реактора в критическое состояние и соответствующей ей скорости изменения концентрации борной кислоты. Структурно комплекс состоит из двух, выполняемых независимо друг от друга программ IST3 и IST4, которые соответствуют двум вариантам проведения расчета:

1) Программа IST3 предназначена для оперативного расчета значений мощности источника нейтронов и параметров средств воздействия на реактивность при пуске реактора после остановки. Под оперативным расчетом понимается расчет, производимый для конкретного пуска после

уже произошедшеи остановки реактора, и в нем учитывается предистория работы реактора перед остановом.

2) Программа IST3 предназначена для упрощенно-консерватиного расчета значений определяемых параметров. Результаты упрощенно-консерватиного расчета, сделанне заранее, магут быть использованы при любом пуске в течение работы данной топливной загрузки. Результаты такого расчета занижены по сравнению с результатами полного оперативного расчета для одинаковых времен работы и стоянки реактора, т.к. не учитывают источник неитронов за счет (γ, n)-реакции.

Допустимая скорость ввода реактивности определяется по зависимости относительной плотности неитронов в момент достижения периода реактора 50 с от скорости ввода реактивности, полученной по результатам расчетов по программе RNMKU и представленнои в комплексе IST, в соответствие с Методикои, вычисляетя по величине мощности источника неитронов в подкритическом реакторе.

При расчете мощности источника неитронов в подкритическом реакторе учитываются следующие процессы, дающие, как отмечено в /2-4/ значимый выход нейтронов:

1) Спонтанное деление изотопов урана и трансурановых элементов, накопившихся в процессе работы реактора на мощности;

2) (α, n) -реакции α-частиц, испускаемых при распаде ядер урана и трансурановых элементов, с ядрами изотопов O^{17} и O^{18} кислорода, содержащегося в UO_2-топливе:

- ${}_8O^{17}$ + ${}_2He^4$ --- ${}_{10}Ne^{20}$ + n + 0.588Мэв;
- ${}_8O^{18}$ + ${}_2He^4$ --- ${}_{10}Ne^{21}$ + n (порог реакции 0.8536Мэв);

3) (γ, n) -реакции на дейтерии, содержащегося в составе воды теплоносителя-замедлителя.

При определении мощности источника нейтронов за счет спонтанного деления и (α, n)-реакции используется апроксимационная зависимость удельной мощности источника нейтронов от выгорания, полученная на основе данных, приведенных в различных работах о нейтроном излучении отработавшего топлива. Спад выхода нейтронов, связанный с распадом нуклидов во время стоянок реактора, учитывается в виде коэффициента, зависящего от календарного и эффективного времени работы реактора.

При определении мощности источника нейтронов за счет (γ, n)-реакций учитываются только высокоэнергетические гамма-кванты, образующиеся продуктами деления с периодом полураспада более часа. Мощность источника по каждому нуклиду определяется в аналитическом виде как решение системы дифференциальных уравнений, описывающих накопление материнских и дочерних ядер в соответствующих цепочках радиоактивного распада. В расчете учитывается предистория работы реактора, т.е. изменение мощности реактора до его остановки, путем разбиения интервала работы реактора на участки с постоянной мощностью и решением системы дифференциальных уравнений для каждого участка.

Мощность источника нейтронов за счет (γ, n)-реакций определяется по мощности источника гамма-квантов с использованием коэффициента пересчета K(γ, n), полученного в результате расчетов по программе РОТОК.

3. Программы РОТОК, предназначенной для определения удельной мощности источника нейтронов в активной зоне за счет (γ, n) -реакций по известной средней удельной мощности гамма-излучения в ТВЭЛ и включает в себя расчет плотности потока гамма-квантов от самопоглощающегося объемного источника с цилиндрической геометрией с учетом ослабления в многослойной защите.

В результате расчета по программе РОТОК определяется значение коэффициента перехода от средней удельной мощности гамма-излучения в топливе к средней удельной мощности источника нейтронов в активной зоне за счет (γ, n)-реакции производится по формуле:

$$K(\gamma, n) = \frac{\Phi \cdot N_{д} \cdot б_{д} \cdot n_{твэл} \cdot S_{кс}^{H_2O} \cdot H}{V_{аз}},$$

где: Φ – средняя по кольцевому объему теплоносителя окружающему ТВЭЛ, удельная плотность поток гамма-квантов;

$б_{д}$ – сечение (γ, n)-реакции на дейтерии;

$N_{д}$ – плотность ядер дейтерия в теплоносителе;

$n_{твэл}$ – количество ТВЭЛ в активной зоне;

$S_{кс}^{H2O}$ – площадь сечения кольцевого слоя теплоносителя;

$V_{АЗ}$ - объем активной зоны.

С учетом консервативного подхода значения допустимых скоростей ввода реактивности рассчитываются для значений CH_3BO_3, соответствующих началу диапазона, в котором производится изменение CH_3BO_3. Примеры представления результатов оперативного и упрощенно-консервативного расчетов приведены в Таб. 1,2 с данными на начало пускового интервала CH_3BO_3. Величины расхода подпитки I контура чистым конденсатом, обеспечивающие соответствующие скорости снижения CH_3BO_3 в I контуре при различных значениях CH_3BO_3 легко определяются по заранее просчитанным зависимостям.

Упрощенно-консервативный расчет целесообразно производить на весь период работы данной топливной загрузки. Данные оперативного расчета (время расчета при условии наличия данных по предистории работы блока ~ 15 мин) заблаговременно представляются персоналу БЩУ с учетом возможности пуска блока в запланированное или произвольное время. Расчеты, проведенные для 4-ой топливной загрузки блока No 1 Хмельницкой АЭС, представлены в Таблице 3. Из таблицы видно, что уже при выгораниях ~ 60 эффективных суток допустимый расход чистого конденсата в пусковом интервале составляет больше 60 т/час, что превышает возможности системы подпитки-продувки.

Применение разработанной Методики позволяет в несколько раз увеличить расход чистого конденсата и соответственно уменьшить время водообмена при выводе реактора в критическое состояние, причем, начиная с некоторого времени работы топливной загрузки, расход чистого конденсата ограничивает уже не допустимая скорость ввода реактивности, а возможности системы подпитки-продувки.

В настоящее время Методика и прогроммные средство внедрены на Хмельницкой АЭС, производится их внедрение на Южно-Украинской АЭС и Российских АЭС по договору с концерном "Росэнергоатом".

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Таблица 1

Результаты расчета по программе "IST3" с примером данных расчета для 3-ей топливной загрузки блока No 3 ЮУ АЭС. Т = 268, эфф. суток. Дата остановки реактора 20.09.92.

Время стоянки	Средняя удельная мощность источника нейтронов			Предельная скорость ввода реактивности	Предельная скорость уменьшения концентрации H_3BO_3 в реакторе
	суммарная	(гамма-n) реакций	спонт. деление и (альфа-n) реакции		(г/кг)/час
сутки	н/см^3 сек	н/(см^3 x сек	н/(см^3 x сек	$\beta_{эфф}$/сек	(г/кг)/час
0.00	5.37e + 02	3.28e + 02	2.10e + 02	1.34e - 02	1.52e + 01
10.00	2.20e + 02	1.63e + 01	2.04e + 02	6.30e - 03	7.13e + 00

Таблица 2

Результаты расчета по программе IST4 с примером данных расчета для 3-ей топливной загрузки блока No 3 ЮУ АЭС

Эффективное время работы 3 топливной загрузки $T_{эфф}$	Продолжительность календарного времени от дня вывода реактор с 3 топл. загрузки на мощность $T_к$	Средняя удельная мощность источника нейтронов в активной зоне S	Отношение средних плотностей нейтронов в акт. зоне при МКУ-мощности и при подкритичности ρ_0= -5$\beta_{эфф}$, пмку/по	Предельная скорость ввода реактивности	Предельная скорость уменьшения концентрации борной кислоты CH_3BO_3 в реакторе (dc/dt) пр.
эфф. сутки	н/см^3 сек	н/см^3 x сек	н/см^3 x сек	$\beta_{эфф}$ /сек	г/кг/час
0.00	0. 00e + 00	2.07e + 01	2.16e + 02	1.34e - 02	1.83e - 00
0.00	8.00e + 01	1.67e + 01	2.68e + 02	1.20e - 03	1.64e - 00
20.00	2.00e + 01	2.66e + 01	1.68e + 02	1.53e - 03	2.05e - 00
20.00	1.00e + 02	2.14e + 01	2.09e + 02	1.36e - 03	1.83e - 00

Таблица 3

Результаты расчета предельных величин скорости ввода положительной реактивности при выводе реактора в критическе оостояние и соответствуЮщих значений скорости уменьшения CH_3BO_3, расхода подпитки I-го контура чистым конденсатом для ВВЭР-1000 блока I Хмельницкой АЭС в период работы 4-ой топливной загрузки.

No пп	Дата остановки реактора	Эффект. время работы	Дата пуска	Длительность стоянки	Средняя удельная мощность источника нейтронов	Величина I-го предела скорости ввода положительной реактивности (Пр. 1)	Величина 2-го предела скорости ввода положительной реактивности (Пр. 2)	Предельные скорости уменьшения CH_3BO_3, соответствуЮщие:		CH_3BO, в начале пускового интервала	Предельные значения расхода чистого конденсата, соответствуЮщие:	
								Пр. 2	Пр. 2		Пр. 1	Пр. 2
					Консер./опер. расчет	Консер./опер. расчет		Консер./опер. расчет			Консер./опер. расчет	
		сутки			н /(см^3/с)	x 10^{-3} $\beta_{эфф}$/с	x 10^{-3} $\beta_{эфф}$/с	(г/кг)/час		г/кг	т/час	
1.	08.04.91 - на перегрузку	0	30.07.91. I-ый пуск с 4-ой загрузкой	113 сут.	14.7 / 14.7	1.12 / 1.12	3.09	1.64 / 1.64	4.52	10.2	32 / 32	88
2.	12.08.91	8.8	20.08.91	8 сут.	14.0 / 22.1	1.10 / 1.45	3.09	1.61 / 2.07	4.41	10.1	32 / 40	88
3.	10.10.91	60.9	13.10.91	3 сут.	27.6 / 46.7	1.56 / 2.17	3.09	2.16 / 3.01	4.29	9.4	44 / 66	93

Development of Informational System of NPP "Kozloduy"

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1. Introduction

The needs of information for the different departments of NPP are defined from their staff performed activities and resolved tasks. Large quantity of information allocated in different departments of the plant is collected during it's construction and operation. A part of this information is related to the equipment and rules of operation and other is received as result of measures and accounts in the process of operation. An approximately estimation of the informational necessities can be done accordingly [1] where the basic functions of informational system are formulated . The precized functions of informational system are enumerated bellow.

2. Functional tasks

After investigations of international requirements to informational systems for NPP and taking into account the particularity of NPP "Kozloduy" the following functional tasks were precized:

1. Ensuring and updating of equipment data.
2. General indicators for operation of NPP.
   - national rule specific indicators.
   - international standard required indicators.
3. Informational support of operator.
4. Neutron physical characteristics and technical parameters of the core accounting support.
5. Data for neutron fuel cycle, input supervision, transportation and preservation of nuclear fuel.
6. Engineer's calculations and analysis.
7. Probabilistic safety analysis
8. Prolongation of operational term.
9. Optimization of regulated verifications of the equipment.
10. Reliability Centered Maintenance
    - Schedule and maintenance task term optimization.
    - Spare parts management.
    - Input supervision of materials and QA.
    - Reliability, availability and maintenance fitness analysis
    - Fault and failure trends.
11. Medical service, individual dosimetric supervision, radilogical situation.
12. External dosimetry, meteorological service, control sensors on radiational situation, civic defence.
13. Administrative financial information - book keeping, cash, main accountant, clerkship, personnel department, pass ,etc.
14. Support of documentation
15. Leading of operative journals and communications of managing staff of NPP with operational and maintenance personal.

3. Data, data bases

The basic data for equipment serve almost all functions of IS and have various applications. After the creation of equipment data base it communicates in on-line mode with all other subsystems and ensures necessary functional subtasks.

Equipment data base encloses:

- constructional descriptions

- ingredient specifications
- technical characteristics
- operational characteristics
- echnologic schemes
- drawings
- other technical and technologic characteristics

The operational data of NPP comprises general functional indicators for operation of the plant and functional indicators for different departments (reactor building, turbine hall, chemistry ,etc.). Specialized software is provided for treatment and storage of necessary data in the different departments and for generation of reports according to accepted in NPP rules. The generating system of general operational indicators for NPP provides automatic manufacture of report documents for regulatory body (CPUAE - Committee for Peaceful Use of Atomic Energy).

The input functional indicators of IS include electrical power, heat power, chemical reagents dose loads, availability indicators of safety systems, etc.

The functional indicator data are received mainly by Unified Computer System (UCS) and their treating is connected with the duties of shift turbine operator, shift reactor operator, technologist operator and shift supervisor. It is used data from chemistry, radiation safety sector, etc. The data represent analogical, descret signals and texts. The following departments of the plant receive data linked with functional indicators and treat them: Control rooms 5 and 6, Production and Technical Department (PTD), Unified Computer System (UCS), Engineer's Support (ES), Radiation Safety and Dosimetry (RSD), Reactor Building (RB) and Turbine Hall (TH).

The Informational System gives possibilities for generation of all required by CPUAE general indicators and documents for operation of the plant including quoted in WANO Performance Indicators /10/92.

The necessary data for informational support of operator include mainly operative information from technologic process of I and II circuitries received by UCS.

The data for informational support of engineer's calculations and analysis include
- data of engineer's calculations of equipment components received by manufacturer
- data of calculations fulfilled in different departments of NPP and external organization.
- delivering of technical and technological data for engineer's analysis and accounts during the modernization of technological schemes, replacement of the part of subsystem and components of equipment.

The necessary data for PSA can be subdivided on data about primary and initial events appearing in the risk model logic (i.e. event trees and fault trees) and data for components which are common for other functional tasks. The necessary data for PSA will be entered by multiple work places and the analysis will be performed in engineer's support department. Since this peculiarity it's recommended development of data base for PSA in ES where by the component code to be linked with the equipment and maintenance data bases.

The specific data necessary for probabilistic estimation are initial events - events linked with components, events linked with human factor and dependant events. For primary events is required the probability of occurrence and for initial events the frequency of their occurrence.

The data linked with the components are two types representing probabilities for primary events: unavailabilty caused by fault and unavailabilty caused by planned repair works or maintenance. For multiple components are required data characterizing specific kinds of faults.

The necessary data for evaluation of human factor are the number of possible human interference, number of errors, activity localization, type of available indication. The dependant event data includes the total duration of the condition in defined period, the duration of that period and specific for initial event information. ALL the data can be received from the logs of significant incidents or their equivalence.

The tasks for prolongation of life operation period are related to critical for operation and safety components:

- residual term of operation
- anticipated sorts of faults, their mechanism, their rate of occurrence
- the cost of repair
- availability of spare components and the cost of replacement

To justify all these requirements except data linked with the components participated in PSA there are needed and data of external conditions exerting influence on the component, transient data for components and systems, etc.

The optimization of regulated verification of equipment requires:

- identification of checked components (from data base of equipment)
- criticality of these components over safety and operation of NPP (data received from PSA)
- data for verifications including: sort of test, conditions, methodology and measuring tools, lap from the last verification, results of verification, sorts of defaults and failures, the reasons for unavailabilty of the component, personification of the people working with this equipment.

The task for Reliability Centered Maintenance represents functional system including several subsystems:

- Optimization of schedule and maintenance tasks terms
- Management of spare parts
- Supervision of input materials and quality assurance
- Reliability, availability and maintenance fitness analysis
- Default trends

The task for preservation and storage of documentation encloses preservation and updating of technological schemes, operational and maintenance instructions, etc. The support of the current documentation should enclose automatic generation of reports and references linked with operation:

- daily, weekly, monthly and annually reports
- communications of the NPP leaders with the operative and maintenance staff
- orders, permissions as well as interdepartmental communications

The task has large range and corresponds with all supported by informational system tasks. It's necessary achievement to the data used by other tasks and any additional data depending on concrete requirements. In general these data can be taken off from existing now international and national standards concerning the operation of NPP.

The list of the tasks can be modified or enlarged depending on concrete conditions, operation experience, necessities of management, existing national lows and international rules.

After analysis of the volume and structures of the data on the individual data bases indicated above it's defined the sorts of relational data bases. Taking in account the necessity for work in multiusers media, the quantity of the users, the reliability for preservation of data and requirement for open type informational system it's accepted it to be developed on the base of operational system UNIX. For development of the data base it's relayed on Informix On line making archive automatically and insuring reliable means against unauthorized access.

The definition of informational system configuration is linked with the organizational structure of the plant and defined DB as well as the analysis of work stations where basically will be performed the functional tasks.

For realization of the informational system is necessary development and connection of several local net (LN) and separate personal computers as emulated terminals to powerful computers in the role of file servers. In the base local net supported the function of the informational system to participate several basic file servers.

The main computers (file servers) should offer enough speedy performance, computing power, large capacitate for conservation of information for stated tasks as well as convenient communication possibilities. The choice of computers for basic file servers ought to be done later at the development of IS. As separate local net servers can be used PC486/33(50). The characteristics of the main computers and terminals are defined by their purpose and volume of the treated information. As PC terminals can be used the existing PC 386/486.

The safety of conserving data will be achieved by use of optical disk fields, and locally by streamers. The quality of documents will be ensured by high resolution laser net printers. The graphical information will be stored in DB in compressed form.

Multi purpose informational system development began in the new part of NPP (unit 5 and 6) by training of system administrators, picking up the data on some functional systems and installing of the system software in local network.

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BG9600407

SAFETY SYSTEMS I&C EQUIPMENT RELIABILITY ANALYSES OF UNIT 3 AND 4 OF NPP "KOZLODUY"

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Risk Engineering Ltd.

1. Purpose of the analyses

The analyses purpose is to assess the safety systems I&C equipment reliability. The assessment includes:

- quantification of the safety systems unavailability due to I&C component failures;
- definition of the minimal cut sets, leading to the analysed safety systems failure;
- quantification of the I&C equipment importance measures of the dominant contribution components.

2. Safety systems functions

The safety systems are designed to prevent accidents and to mitigate accident consequences. They have to perform defined functions in case of accident initiating events. Some of the more important safety systems functions are:

- reactor trip by means of inserting hard absorbents in the core;
- maintain the core in sub critical state by means of inserting liquid absorbents in the primary circuit;
- core residual heat removal in case of high primary circuit pressure;
- core residual heat removal in case of low primary circuit pressure;
- maintain the primary circuit inventory in case of high primary circuit pressure;
- maintain the primary circuit inventory in case of low primary circuit pressure;
- avoid a confinement over pressure in case of LOCA.

3. Safety systems I&C equipment

I&C equipment supports the safety systems to perform the safety functions. I&C equipment automatically starts the safety systems in case of accident initiating event. In general the I&C equipment perform:

- control the normal operation processes;

- front line and support systems equipment protections and interlocks;

- normal operation systems start and remote control;

- control the core and heat transfer processes;

- indication in case of critical parameters;- actuating and control the front line, support and mitigation systems during performance of the safety functions.

To provide higher reliability, units 3 and 4 safety systems I&C equipment is designed as follows:

the control systems, the reactor protection system and the interlocks are designed in three independent channels generally;

majored channels actuation logic of the protection systems and interlocks(2 out of 3, 2 out of 4, 3 out of 6);

the independence of the channels is provided by: autonomous control equipment, automatic and remote control, primary converters and I&C equipment for every channel of the system, independent power supply for the instrumentation channels without interconnections between them, physical separation between the channels.

4. Safety systems I&C equipment reliability analyses general methodology

The safety systems I&C reliability analyses methodology consists in three general steps: choice of reliability parameters; development of mathematical models; definition of the reliability assessment criteria.

To be adequate to the real system, the methodology have to consider the specific properties of the system:

- functions the system have to perform;

- system structure concerning every function that have to be performed;

- performance of the system and its components;

- maintenance and repair conditions;

- failure modes and consequences of the system and its components.

The safety systems I&C equipment assessment consists in performing quantitative and qualitative analyses of the safety systems reliability due to the I&C equipment failures. It includes an initiating event analyses, requiring systems functioning. Initiating events with the same safety systems requiring, have to be united in groups. The total frequency have to be defined for every accident initiating event group. Safety systems hardware analysis have to be performed. A definition for channel and system failure have to be completed. Channel failure number leading to system failure have to be defined.

The failure modes, the detection and recovery conditions for every component have to be analysed. The quantitative reliability parameters like failure rate, mean repair time and so on have to be determined.

The analyses results have to be input in tables and channel reliability logic diagrams have to be compound for all the systems, considering the analysed group of accident initiating events. Fault tree (FT) have to be used as reliability diagram. It express all the basic events (component failures) or combinations of them that will lead to the final event of the analysed system (system failure). Usually general and reduced fault tree is build. The reduced fault tree is obtained from the general fault tree removing component failures with negligible contribution to the system failure probability. All the system component failures are removed with the purpose to assess the support I&C component failure contribution.

5. Fault tree general procedure

The fault tree construction procedure [1] generally includes the next steps:

- define the failure mode of the analysed system;

- precise investigation of the possible states and the supposed operating modes of the system;

- define the functional properties of every basic event and the impact on the system;

- fault tree construction for a logically related basic events. Input failure probability, unavailability coefficient, failure rate, repair rate and another parameters, characterising the basic events.

- fault tree qualitative assessment;

- fault tree quantification;

- I&C components importance assessment.

PSAPACK 4.2 was used for fault tree construction, minimal cut sets definition, quantification and assessment of the importance measures of the I&C components. In Appendix 1 is printed the emergency feed water system fault tree. In Appendix 2 are printed the minimal cut sets of the emergency feed water system I&C equipment.

6. Analysed systems

Following the described methodology the next safety systems were analysed:

- neutron [illegible] control equipment analysis;

- reactor protection system analysis;

- main coolant pumps I&C equipment analysis;

- safety systems staggered load system analysis;

- analyses of I&C equipment of the next systems:

  pressuriser safety valves, type "Sempell", of the over pressure protection system of the primary circuit;

  steam dump to the atmosphere and steam dump to the condenser valves;

  spray system;

  low pressure injection system;

  emergency feed water system;

  essential service water system.

Three reports were issued containing the considered systems analyses and results. The pressuriser safety valves "Sempell" analysis of units 1- 4, the safety systems analyses of units 3 and 4 are in the first and second volumes. The I&C equipment analyses of units 3 and 4 are in the third volume.

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Анализ надежности КИП и А систем безопасности пятого и шестого блока АЭС "Козлодуй"

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Риск Инжинеринг ООД

1. Цели анализа

Целью проведенного анализа надежности систем КИП и А является определение областей ненадежности, на которых можно влиять, т. е. найти тех областей, где изменением проэкта или изменением режима испитаний возможно повысить общую надежность систем управления.

Другую цель анализа является получение качественной и количественной оценки влияния неготовности автоматики на неготовностью систем безопасности.

Объем исследования ограничен расмотрением управления спринклерной системы (TQn1), системы аварийного впрыска низкого давления (TQn2), системы аварийного ввода борного раствора (TQn3), системы гидроаккумуляторов (YT), ИПК компенсатора давления; системы сжатого воздуха и газовые сдувки (YR), системы сжатого воздуха для пневмоприводов локализирующей арматуры (UT), ИПУ ПГ, БРУ-А, БЗОК,

Анализ надежности систем КИП и А проведен использованием метода дерево отказов. Для количественной оценки использован программный пакет "PSAPACK", версия 4. 2.

2. Подход при моделировании

Анализ управления данной технологической системы связан с функциями и действиями, которые система выполняет автомати-чески.

Объект исследования является успех автоматики привести систему в состоянии, необходимым для выполнения ее функций.

В анализе принимается, что исходное состояние систем, управление которых исследуется, соответствует определенному в инструкции эксплуатации данных систем. Поэтому рассмотриваются только сигналы и команды, направленные к активным элементам, изменяющие свое положение при работе системы по предназначению.

Запреты и команды, которые не приводят к реальному изменению положения элементов технологической системы не рассматриваются.

Отказ системы управления дефинируется как *неуспех елементов КИП и А привести технологичесную систему в работоспособном состоянии при наличии запроса* (т.е. после срабатывания АСП) *или параметров пуска.*

Так свормулированный критерий успеха упрощает модель отказов в рамках, прилежащей к системе автоматики, т.е. к эле-ментам управления, уникальных для данной системы, ограниченных системным шкафом управления. Это позволяет идентифицировать отказы в цепях системы, а отказы в управляющих цепях, общих для всех систем безопасности (элементы АСП и общеблочных защит) рассматривать в отдельном исследовании. Полученные результаты будут коректными по отношению к неготовности системного управления, избежая лишнее усложнение модели.

Известно, что системы управления являются источником зависимых отказов. Воспринятый в исследовании подхода дает возможность детального рассмотрения сигналов в отдельных местах их формирования с целью прецизного определения областей, генерирующих таких зависимостей.

Модели построены для одного канала систем. Симетричность управления многоканальных систем позволяет полученные результаты для одного канала обобщить и на других. В случае несиметричности строится модель для каждого отдельного канала систем управления.

Модель систем автоматического управления строится для каждого сигнала, следуя его структурную схему.

Отказ сигналов моделируется в терминах отказов составляющих его элементов КИП и А.

Особая проблема при построении модели систем КИП и А представляет определение вида отказа логических блоков типа БЛП, БЛВ, БПН и др. Характерное свойство этих элементов является, присущая им многоканалность. Они имеют независимые входы и выходы, что позволяет прохождение разных сигналов, направленых к разным элементам. Это предполагает в ДО моделировать отказы на уровне составных элементов БЛП, БЛВ и др., отдельных входов и выходов. Имеющиеся литературные данны для отказов оборудования этого типа не позволяют такую детализацию. В анализе отказы этих элементов дефинируются, как отказ блока функционировать в целом. Внесенный таким образом известный консерватизм в оценки надежности системного управления учитывается при анализе результатов.

3. Результаты

В исследовании надежности КИП и А были получены следующие количественные оценки для неготовности систем безопасности выполнить свои функции вследствии неготовности автоматики. Представленые ниже результаты относятся к одному каналу системы.

Система TQn1:

- неготовность канала системы управления 6.0484E-2

Система TQn2:

- неготовность канала системы управления 1.2259E-2

Система TQn3:

- неготовность канала системы управления 3.6981E-2

Система TQn4:

- неготовность канала системы управления 1.8742E-2

Система YT:

- неготовность канала системы управления 3.8138E-3

ИПУ компенсатора давления

- неготовность управлениея 9.2580E-12

Система YR:

- неготовность канала системы управления на линии, связанной с рактором (YC) 3.2583E-5
- неготовность канала системы управления на линии, связанной с компенсатором давления (YP) 3.2583E-5
- неготовность канала системы управления на на линии, связанной с парогенератором (YB) 1.0523E-3

Система UT:

- неготовность канала системы управления 5.9943E-3

Импульсно- предохранительное устройство ПГ

- неготовность управления 1 4917E-4

БРУ-А

- неготовность управления 1.850E-2

БЗОК

- неготовность управления 3.9843E-5

4. Анализ результатов

Полученные минимальные критические группы сечений (см. выборку результатов) дают качественную оценку надежности систем КИП и А .

Исходя из результатов всех систем следует, что особое внимание необходимо уделить на повышению надежности электропитания КИП и А оборудования, а точнее на питание датчиков и шкафов УКТС.

Полученные результаты идентифицируют особую группу элементов, каждый из которой представляет минимальную критическую группу I порядка, но имеет высокие показатели надежности. Это БУД, БУЗ напорной арматуры, соответствующие БКЛ и последовательно связанные логические блоки. Вклад этих елементов в общую неготовность систем самый большой из всех КИП и А елементов. Это лудше всего видно при просмотре результатов ДО для YR, БРУ-А. Один из возможных путей повышения надежности этих элементов есть изменение в режиме их проверок (особенно для тех, которых подлежат тесту раз в году). Для сравнения неготовность БУЗ есть 7.21E-4 при регламентированных проверок раз в месяц и 7.17E-3 при режиме проверок раз в году.

Сравнительно большие значения, полученные для неготовности систем безопасности, обусловленной отказам автоматики, являются результатом огромного количества сечении, полученных при обработки ДО. Необходимо отметить, что отдельные группы сечений имеют, как было сказано, высокие показатели надежности, а также что результаты относятся к надежности одного канала системы.

В итоге на основании полученных результатов можно обобщить, что управление систем безопасности АЭС "Козлодуй" имеет высокую надеждность.

Выборка полученных результатов анализа надежности КИП и А систем для АЭС "Козлодуй" 5,6 блока.

Представленна часть результатов, полученных для систем TQ12, TQ13 и для одного БРУ-А. Выборка содержить только сечений I порядка и часть II порядка

MINIMAL CUT SETS Date: 05/17/94 Time: 08:15:05

<<< P S A P A C K V. 4.2 >>> *GTQ12-P* *

Average probability = 1.225940e-002

1 2.796084e-003 KLV047-F

2 1.608705e-003 LV047-F

3 1.608705e-003 LV04-F

4 1.608705e-003 BV-F

5 9.738548e-004 TQ12D01Y-FO

6 9.017608e-004 TQ12D01L-FO

7 9.017608e-004 TQ12S04L-FO

8 7.214957e-004 TQ12S04Y-FO

9 4.618933e-004 HV42BZ

10 1.079922e-004 HV42/13L-FO

11 1.079922e-004 HV42/12L-FO

MINIMAL CUT SETS Date: 05/17/94 Time: 08:16:44

<<< P S A P A C K V. 4.2 >>> *GTQ13* *

Average probability = 3.638190e-002

1	4.570596e-003	HL01BZ2	HL01BZ
2	4.570596e-003	HL01BZ1	HL01BZ
3	4.570596e-003	HL01BZ1	HL01BZ2
4	2.796084e-003	KLV025-F	
5	2.796084e-003	KLV034-F	
6	2.586649e-003	LV025-F	
7	2.586649e-003	LV034-F	
8	1.608705e-003	LV02-F	
9	1.608705e-003	LV03-F	
10	1.608705e-003	BV-F	
11	9.017608e-004	TQ13S07L-FO	
12	9.017608e-004	TQ13S26L-FO	
13	8.994602e-004	TQ13D01L-FO	
14	7.214957e-004	TQ13S07Y-FO	
15	7.214957e-004	TQ13S26Y-FO	
16	5.757789e-004	HV40/8V-FO	
17	4.618933e-004	HV37BZ	
18	4.618933e-004	HV40BZ	
19	2.700000e-004	TQ13D01Y-FO	
20	1.299206e-004	HV40/34R-FO	
21	1.079922e-004	HV40/20L-FO	
22	1.079922e-004	HV40/21L-FO	

MINIMAL CUT SETS Date: 05/17/94 Time: 07:58:32

<<< P S A P A C K V. 4.2 >>> *GTX50S05* *

Average probability = 1.850856e-002

1	8.946242e-003	TX50S05L-FO		
2	7.165564e-003	TX50S05Y-FO		
3	1.608705e-003	LE01/6		
4	4.618933e-004	HX47BZ		
5	1.984703e-004	EE	BX-CX02-F	
6	6.622160e-005	EE03/A12/2	BX-CX02-F	
7	2.659353e-005	EE	BV28-F	
8	8.873197e-006	EE03/A12/2	BV28-F	
9	8.208413e-006	CX02/8G	EE	
10	4.722646e-006	EE	BX-F	
11	4.722646e-006	EE	CX02-F	
12	2.738820e-006	EE03/A12/2	CX02/8G	
13	1.575758e-006	EE03/A12/2	BX-F	
14	1.575758e-006	EE03/A12/2	CX02-F	
15	1.204469e-006	BX-F	P7/3	
16	9.594593e-007	HG31/A14	HG31/A13	
17	1.733225e-007	BX-CX02-F	P7/3	BU15
18	1.238625e-007	BX-CX02-F	P7/3	B16/3

BG9600409

Reliability analysis of the reconstructed safety systems for unit II of NPP Kozloduy with reactor VVER 440/V230

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Abstract

For the improvement of the safety of NPP Kozloduy unit II with reactor VVER 440/ V 230, reliability analyses are performed of the safety systems of this unit. On the basis of these analyses technical decisions are proposed for reconstruction. The comparative analyses of the safety systems reliability show that the systems reliability is increasing after reconstruction. A PSA for this unit is performed for the case of design basis accident.

The purpose of the Programme [1] developed in 1991 was to increase the safety of unit II up to acceptance level by upgrading of the safety systems reliability.

This unit started operations in 1975 and this type of reactors was designed in the middle 60-ies, according to the Russian design standards and requirements of that time. This causes the need to reconstruct parts of these units in order to prolong their operational life up to the design value of 30 years, in agreement with the increased safety requirements in Bulgaria and other states.

Front line systems ensuring nuclear safety are: High Pressure Safety Injection System, Spray System, Auxiliary Feed Water System, and therefore reliability analyses of these systems are performed.

The approach taken in all safety systems studies the fault tree methodology [2] to determine the unavailability of each system.

Common mode failures are considered for the pumps and valves using the beta factor method [3].

The mission time for each system is 24 hours and the test period is 720 hours.

Support systems are included in the fault tree and, where necessary, human errors are considered also.

All the systems control and instrumentation signals are modelled explicitly in the fault trees. The generic IDEA reliability data base [4] is used for all quantifications.

In order to perform fault tree analyses for a system it is necessary that the success criteria for the system be defined as a result of the event tree. The approach taken is therefore to present the initiating events that would require the system operation and on this basis to determine the thermohydraulic analysis success criteria for each system. The computer code PSAPACK [5] is used for these analyses.

High Pressure Safety Injection System (HPSIS)

The High Pressure Safety Injection System provides emergency injection of boric acid solution to the primary system in case of a loss of coolant accident (LOCA) and ensures the unit nuclear safety. The HPIS principal technological layout is shown on Fig.1.

The analyses are performed with the following main assumptions.

a) The HPSIS is analysed for the initiating event - Design Basis Accident (DBA) LOCA with equivalent diameter Dy 32 mm.

b) Success criteria is: at least one HPSIS pump injects water into the primary system.

Based on these analyses the following technical decisions are proposed:

- To duplicate the motor operated valves AP-5 with a parallel line
- Removing the interlocks for valves AP-5 opening on valves AP-11dr closure.
- The valves AP4 are kept open when the system is in hot standby state.
- Modification of the pump interlock key design.

The results for system unavailability's for the cases before and after reconstruction of HPSIS are given in Table 1. The main contributors to the system unavailability failures before reconstruction are common mode failures and single failures of motor operated valves.

After reconstruction the main contributors are failures of pumps and emergency electrical supply system.

Spray System

The spray system (SS) is designed to cool and depressurize the confinement by spraying water from the emergency borated water tank (EWT) to condense the steam released in a loss of coolant accident (LOCA).

The system technological layout is shown in fig. 2.

The analyses are performed with the following main assumptions:

1) The Spray System is analysed for the initiation event - DBA

2) The success criteria is at least two spray system pumps deliver water from EWT to the confinement.

Based of these analyses the following tecnnical decisions are proposed:

- All three pumps start automatically upon signal of confinement high pressure
- Actuation of valves 1 - 3 SS5 upon signal of confinement high pressure
- To ensure a permanent flow through the spray system heat exchangers, duplicate the motor operated valve SS-6RC with parallel line including normally open valves in parallel (fig. 3)
- Relay redundancy in Instrumentation and control systems

The results for the system unavailability's for the cases before and after reconstruction of the Spray System are given in Table 1.

The main contributors to system unavailability before reconstruction are the failures of the pumps to start and run, unavailability due to testing and failures of the valves SS 6RC.

After reconstruction the main contributors are failures of the emergency electrical supply system.

Auxiliary Feed Water System

The Auxiliary Feed Water System (AFWS) is intended to supply feed water to the Steam Generators (SG) and thus to provide unit cooldown in case of Loss of Main Feedwater. The system technological layout is shown in Fig. 4

The analyses are performed on the following main assumptions:

1. The AFWS is analysed for the initiating event - Loss of main feedwater

2. The success criteria is one AFWS pump feeding at least two SG.

Based of these analyses the following technical decisions are proposed:

- Both AFWS pumps automatically start simultaneously upon signals - Low main feedwater discharge pressure and loss-of-offsite power
- Remove interlocking switches
- Connect the AFWS discharge collector of units one and two and modification on the connection between main feedwater collector and auxiliary feedwater pumps headers.

The results for system unavailability's for the cases before and after reconstruction's AFWS are given in Table 1.

The main contributors to the system unavailability before reconstruction are common mode failures of both AFW-pumps to start and of I&C system.

The main contributors after the reconstruction are human error to open the valves on the piping between the two units.

A probabilistic safety assessment was performed to evaluate the influence of the reconstructed systems on unit II safety. This PSA was performed using event tree methods for initiating event DBA. The safety functions for this accident are given in Table 2 and event tree is shown in figure 5.

The analysis is performed on the following assumptions:

1. Blackout occurs simultaneously with the DBA

2. The reactor protection system scrams the reactor

3. Probability of initiating event - $2.3*10^{-2}$/RY

4. Core damage occurs when cladding temperature reaches 1200°C

5. Unavailability of safety systems is:

Reactor protection system - $2.4x10^{-5}$

Emergency electrical supply system second category - $2.56x10^{-3}$

Steam dump to the atmosphere - fail to open $1.84x10^{-3}$

The analysis of the results shows, that increases safety of unit II after the reconstruction of the safety systems, which a corresponding decrease of the core damage frequency from DBA (from $3.53E^{-3}$, 1/RY before the reconstruction of the safety systems to $1.07E^{-4}$, 1/RY after reconstruction).

Conclusion

Reliability analyses of front line systems of NPP Kozloduy unit II consider proposed decisions for reconstruction. The results show that the reliability of these systems is increasing after reconstructing and that the safety of unit II is upgraded, which decrease the core damage frequency by better than one order of magnitude.

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Table 1 The safety systems unavailability's before and after reconstruction

No	Safety Systems	System unavailability before reconstruction	System unavailability after reconstruction
1	High Pressure Safety Injection System	6.36×10^{-3}	1.47×10^{-4}
2	Spray System	7.78×10^{-2}	3.71×10^{-3}
3	Auxiliary Feed Water System	3.22×10^{-3}	1.03×10^{-5}

Table 2 Safety Function on DBA for unit 2 NPP Kozloduy

No	Safety Function	Safety System	Success criteria
1	Reactor shutdown	Reactor protection system	Putting in the core all control assemblies without one
2	Reactor coolant inventory maintaining	HPSIS	Operation of one HPSIS pump
3	Decay heat removal on high pressure	AFWS Steam Dump to the atmosphere	Operation of one AFWS pump, one SDA
4	Electrical supply busbar by DG	Emergency electrical supply system	Operation of one Diesel Generator
5	Reactor coolant inventory maintaining and decay heat removal on low pressure	HPSIS	Operation of one HPSIS pump
6	Cooling of confinement	Spray system	Operation of two SS pumps

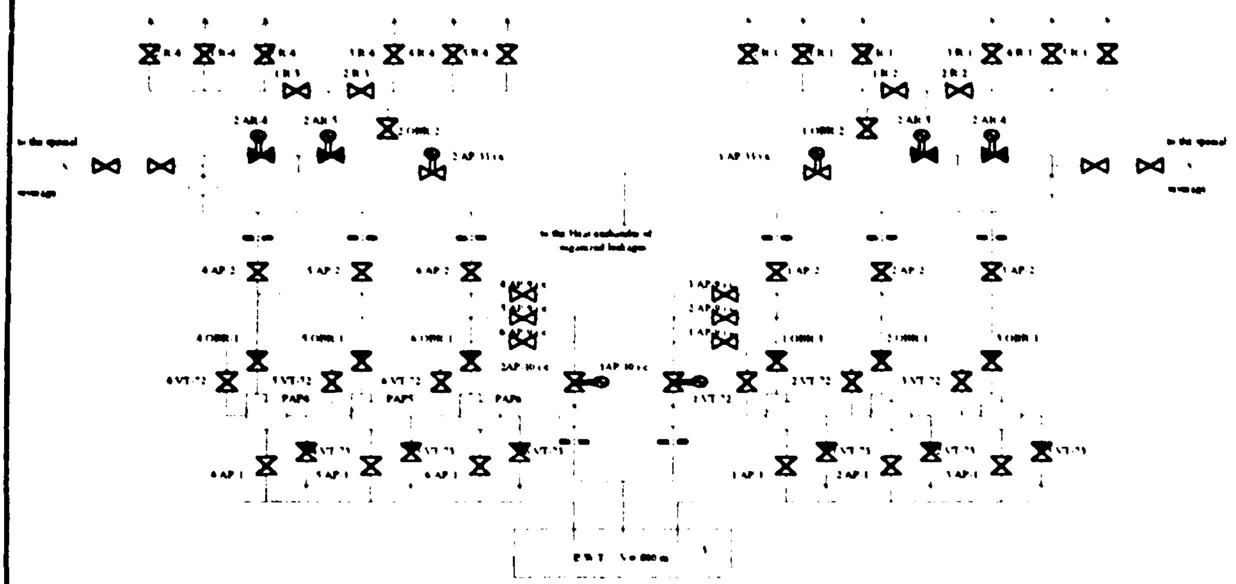

Figure 1
High pressure safety injection system - Principal Technological Layout

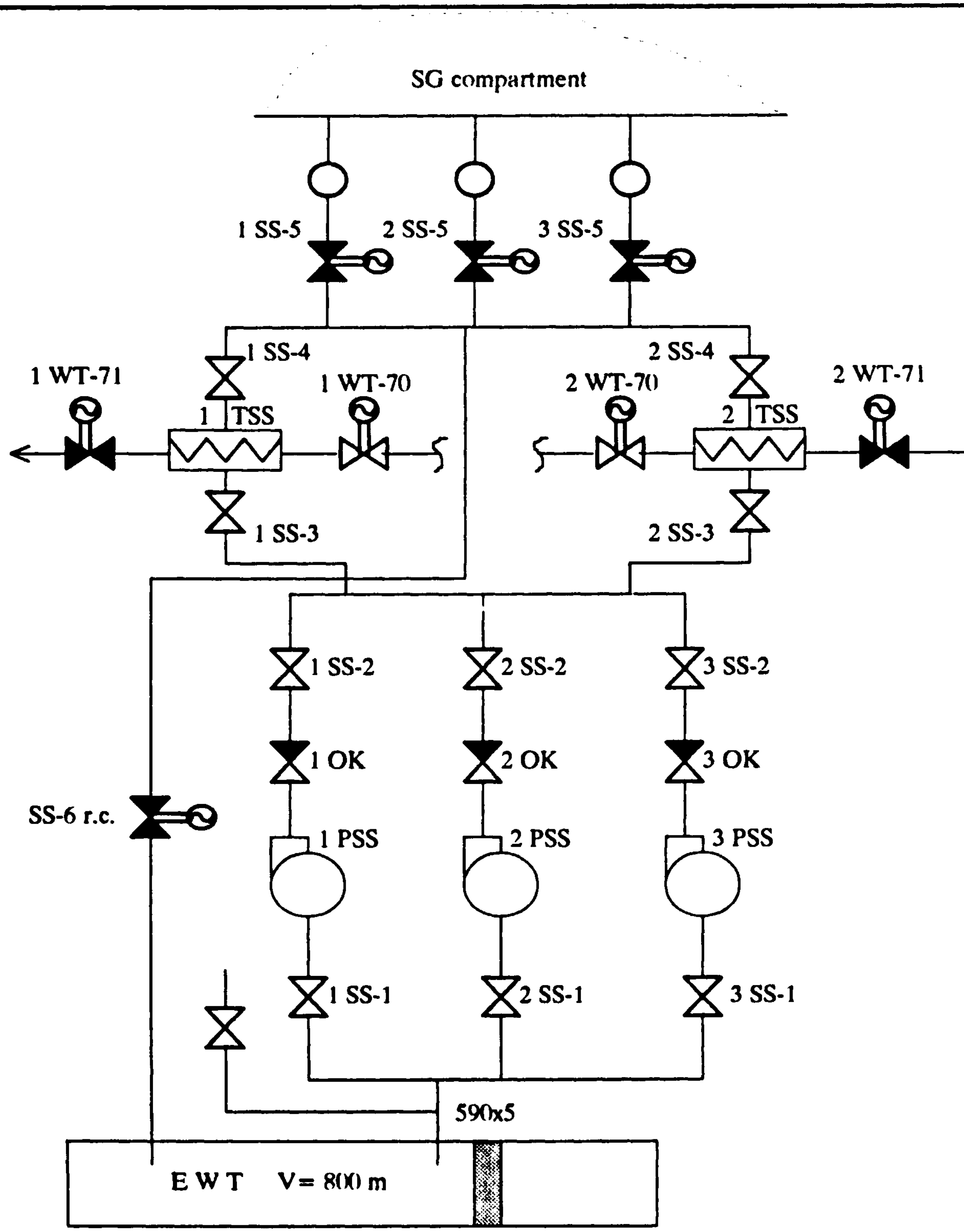

Figure 2
Spray system - Principal Technological Layout

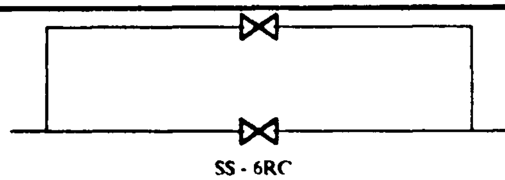

Figure 3

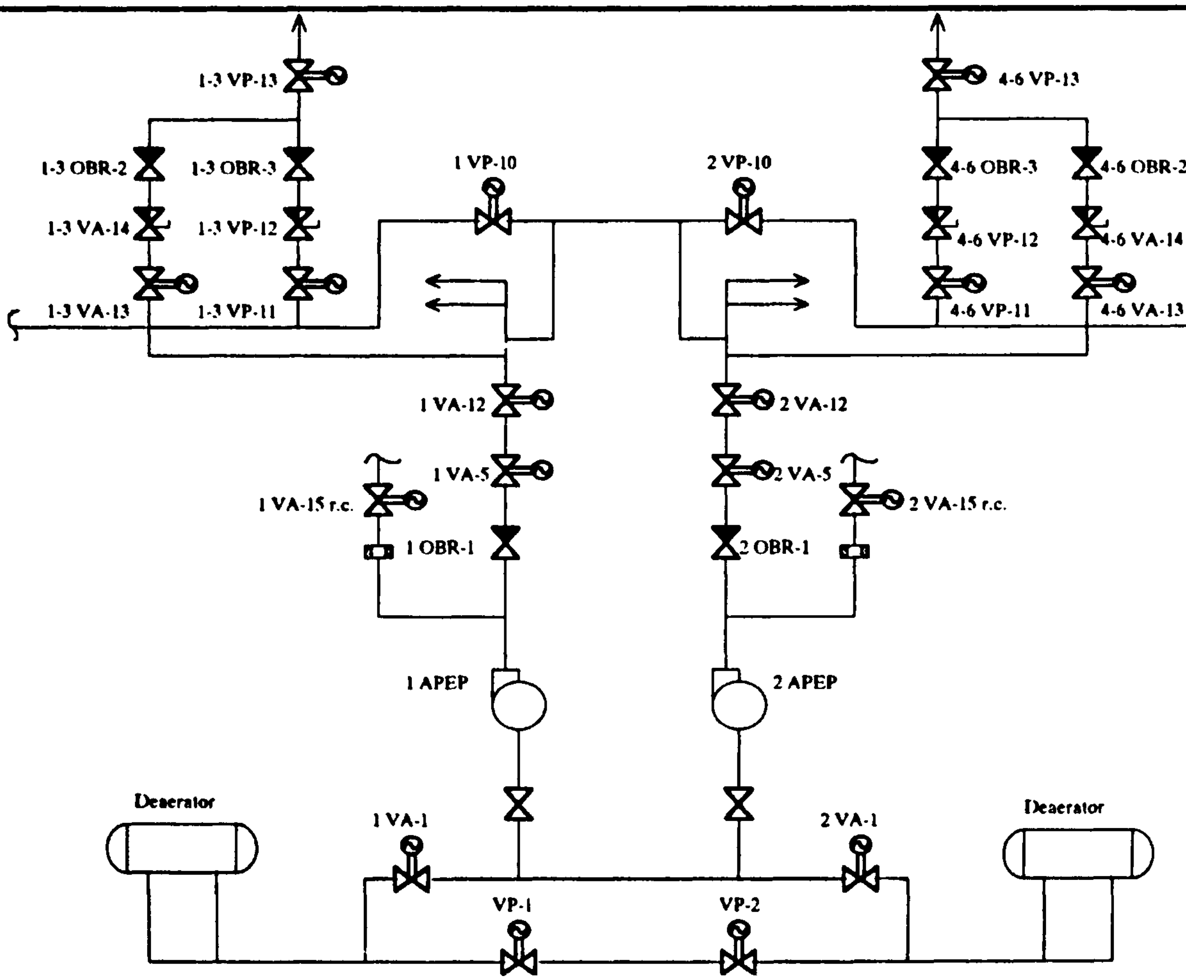

Figure 4
AFWS - Principal Technological Layout

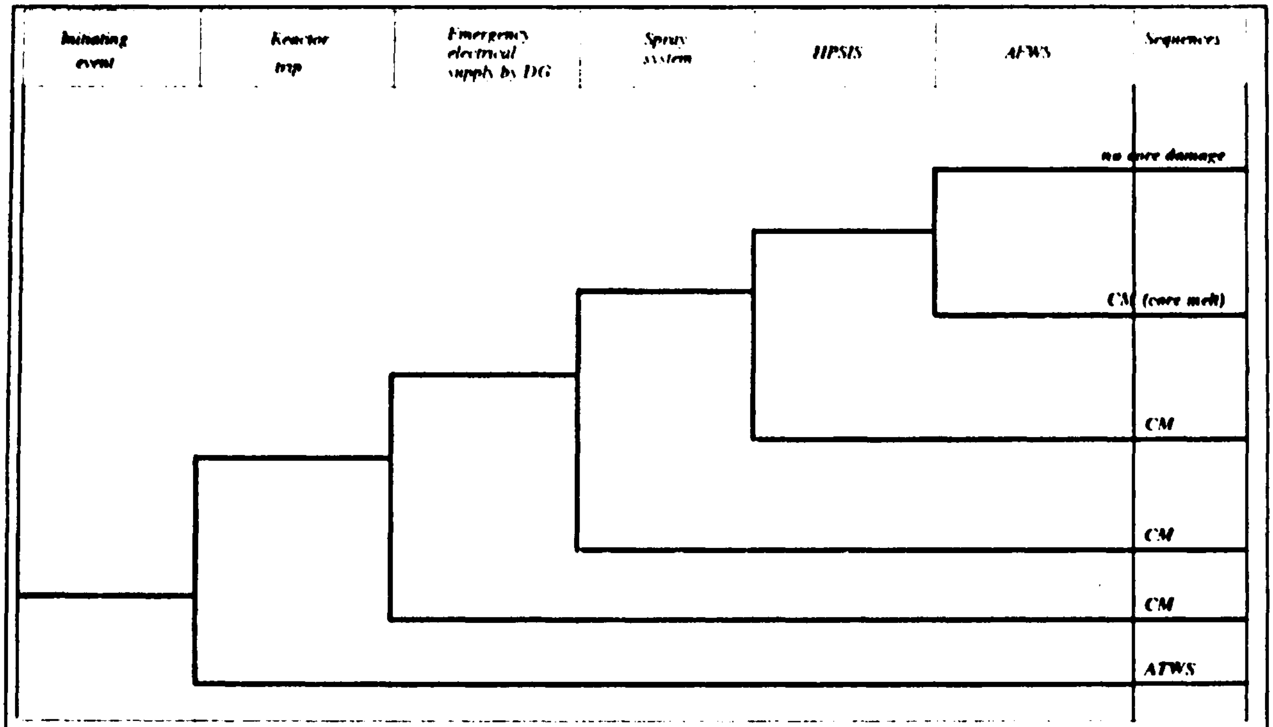

Figure 5
Event tree for DBA with Diametr 32 mm

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Experience based on the researches of systems and events on the units 1-4 of NPP "Kozloduy" from the reliability viewpoint.

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SUMMARY

On the basis of the performed analyses were made useful conclusions and recommendations about measurements for increasing the safety and the ability of the equipments for better task performance.
After the overview of the full volume of the information in the accident protocols of NPP "Kozloduy"-units 1-4 we can make the conclusion, that in the NPP "Kozloduy" in this period there were not occurrences of accidents with consequences for the fuel integrity or radioactive releases outsides of the plant.
The percent of human-caused errors is relative big, more than 20%, but we can make the conclusion, that the personnel of the plant works good and with competence as in more simple so in more complex situations, is gut educated and high professionally, that prevented till now occurrences of heavy accidents in the NPP "Kozloduy".
The reliability characteristics of the equipment or events of NPP "Kozloduy" are similar or identical with these of plants with pressure water reactors.
With not much effort could be increased the reliability of the systems with reduction of the human-caused errors and of the occurrences of error signals.
Much more invest is necessary for eventually reconstruction and modernisation because of the needed variation reliability analyses, new projects, tests and proving the compatibility of the new projects with all other systems of the NPP "Kozloduy"

1. Introduction

Between 1992 and 1994 our team researched the reliability of different equipments of some safety related systems in the NPP "Kozloduy". We analysed on different ways also all recorded protocols about occurred events (serious failures, incidents and accidents) about frequencies, reliability characteristics and qualitative/quantitative relationships [9,10]. Some conclusions from these analyses we try to expose in this paper. Because of the great value of the analysed information, we will explain only restricted part from the made analyses and conclusions, based on the exploitation data from the NPP "Kozloduy".

2. Overview of the reliability of the equipments from some safety related systems in the NPP "Kozloduy". Service water system, Feed water system, Emergency power supply -category 2, Emergency high pressure ejection system, Spray system.

We received exploitation data from these systems about failures of some equipments as pumps and valves. The data about the failures of the equipments concerns the period after 1987 and the involved reliability characteristics are typical for this period.

2.1. Conclusions concerning some reliability characteristics of the pumps of the service water system.

The relation between electrical and mechanical initiators for failures is around 1:28. The most frequent causes for failures are wearing, contamination and other similar mechanical causes. The defects of the equipments , caused failures, in our failure data bank are not very heavy. By the overview of the accident protocols we detected six abnormal exploitation conditions because of loss of service water system's functionality with failure rate λ=4.8.10-5, 1/h [10].
The distribution functions of the failures of the service water pumps have exponential character that indicates the independence between the failures.
The failure rate for occurrence of a failure on a service water pump is in the area

$\lambda \in (3.0.10\text{-}4\text{--}4.0.10\text{-}4)$, 1/h, assessed as pessimistically value that means - there are token into account also the small defects that are repaired lather and the equipment was not stopped immediately.
For the other pumps in the NPP "Kozloduy" is typical a λ value like $\lambda = 2.0.10\text{-}4$, 1/h. Based on the analyses we made the conclusion that it is necessary to be used more clear water for the service water system.
For the group service water pumps from unit 2 were registered two time worse reliability characteristics for the period after 1991. The worse characteristics of this pumps may depend from the bad characteristic of service water pump N5, where we suppose some ageing processes. We can give the followed values of λ for a service water pump from units 1 and 2 as

- for unit 1: $\lambda \in (4.0.10\text{-}4\text{--}5.0.10\text{-}4)$, 1/h;
- for unit 2: $\lambda \in (3.0.10\text{-}4\text{--}4.0.10\text{-}4)$, 1/h.

2.2. Some data from the analysis of the emergency feed water system: feed water pumps and some other components.

Higher frequency of the failures of the emergency feed water pumps was observed by the pumps from units 1 and 3 , but assessed on restricted number of data, related to the period after 1988.
The relation between the failures root causes electrical:mechanical is 1:20. The most frequency causes for failures of these pumps are vibrations, wearing, leaks from sealings and so on.
The human-caused failures are around 14% in the collected data in our information system. The mean repair time for emergency feed water pump is around 16.47 h. and this value is relative big, if we take into account that a failure of a feed water pump decreases the safety of the unit.
The characteristic λ for a emergency feed water pump is between

$\lambda \in (1.6.10\text{-}4\text{--}2.4.10\text{-}4)$, 1/h.

The distribution functions of the failures of this kind of pumps are exponential, that is the best case from the view point of the reliability. For the emergency feed water pump N1 was detected some increasing of the failure rate during the time, but assessed on not enough data.
Concerning other components of the feed water system: some regulators and control valves we can make also some conclusions. We observed some qualitative and quantitative arguments that points to presence of ageing processes.
The human-caused failures take around 20% in our failure data bank.
The data about the different valves is not enough for calculation of the reliability characteristics for every valve. But for example the failure rate for the level regulator in the steam generator is calculated as $\lambda = 2.5.10\text{-}5$, 1/h on the basis of the failure data in our data bank.

2.3. Emergency power supply AC - category 2.

The conclusions about the reliability were made on the basis of the received by us data about the diesel-generators.
Here the relation between electrical:mechanical causes for failures is around 1:1. The root causes with most frequency are:

- vibrations;
- defects of the used material;
- contamination;
- ageing and wearing of elements.

The human-caused rate is around 28%. The human-caused failure rate for the diesel generator is

$\lambda = 4.27.10\text{-}5$, 1/h.

Here was also made the conclusion that the elements with bad construction must be changed ând the cooling water must be more clear.
Based on the received data about the failures we can made also the conclusion, that the failures of the diesel-generators are not very heavy and the mean repairing time is not big: around 4 h.
The pessimistic assessment of the failure rate for a DG lies between

$\lambda \in (1.0.10\text{-}4\text{--}3.0.10\text{-}4)$, 1/h.

The reliability characteristics of the mechanical caused failures are worse comparing to the electrical caused. The failure rate λ is worst for the DG from unit N3 comparing to units 1 and 2.
The most often recommendations to the Diesel-generator system are:

- on time changing of the old elements;
- eliminating the elements with bad construction;
- maintenance optimisation.

2.4. Emergency high pressure ejection system-units 1 and 2, NPP "Kozloduy".

The failures of the equipments of this system are mostly mechanical caused- with greatest frequency- wearing of sealings.
The human-caused failures of the high pressure ejection pumps take around 15%. The data about the failure of the pumps from this system concerns the period 1989-1992.
If we ignore the common-caused failures, we calculated the following failure rates for pumps from this system:

- Failure rate for high pressure ejection pump from unit 1: $\lambda=4.54.10\text{-}5$, 1/h;
- Failure rate for high pressure ejection pump from unit 2: $\lambda=2.31.10\text{-}5$, 1/h.

The emergency high pressure ejection pumps are different for unit 1 and 2. The pumps from unit 1 are type ЕП-50 and from unit 2 -ЦН 65-130. As we can see from the failure rate value λ, the pumps from unit 1 have around two time worse reliability comparing to the pumps from this system from unit 2. Also the mean repairing times for the HPEP pumps from unit 1 is much more greater than for the HPEP pumps from unit 2.
Here is very interesting the fact, that the failure rates we calculated for this kind of pumps is practically identical with the λ values for the same characteristics of the pumps from NPP "Greifswald".
These failure rates are pessimistic assessment of the failures of the HPEP.
On the basis, that the pessimistic assessed failure rate is one degree worse comparing to the optimistically one (takes into account only the heavy failures), we can make the conclusion, that our assessment of the λ is similar to the values, given in [14] about the λ for the U.S.'s PWR.

2.5. Spray system.

The data we received is not enough for complete assessment of the reliability characteristics of the equipments of the spray system. There were analysed some pumps from the spray system on the basis of data about the period 1989-1992.
The received data about the failures of the spray system's pumps includes most mechanical caused failures:wearing, contamination, leaks from sealings. This type of failures are not very heavy and the calculated reliability characteristics are pessimistic. The mean repairing time is around 1 h., that confirms our conclusion given above.
The failure rate for spray system's pups is $\lambda \approx 5.10\text{-}4$, 1/h. This value is one degree worse comparing to the HPEP (boration pumps) probably because of very big rate small defects token into account by the calculation of the reliability characteristics.

3. Experiance based on the analyses of the accident protocols of units 1-4 of NPP "Kozloduy".

These analyses are based on the information in all accident protocols for units 1-4, NPP

"Kozloduy" for the period 1974-1993.
There were overviewed [10] also different documents and conclusions of some international missions [1,12], other information from papers and publications [2, 3, 4, 5].
It was performed a list with 59 initiating events: transients, accidents and defects in some systems with help functions [10]. For such kind of events were calculated frequencies of occurrences, some reliability characteristics and qualitative/quantitative relationships.
Typical initiators for accidents/incidents were researched. We performed also a list with the 39 most often root causes for accident situations. There were calculated their frequencies and percent rate [10]. We made two-step structuring on the basis of each protocol with the goal: easier and better analysis of the events. At the first, we created for each protocol a table-structured representation and after that the event tree for the events sequences in each protocol.

The most frequent human-independent root causes for accidents/incidents are:

- bad isolation-19;
- error signals-19;
- bad electrical contacts-21;
- breaks and leaks from bearings-13;
- Failures during tests or repairing works by the full power of the unit-24 /here not all occurred failures are objective/;
- short circuits caused switch on/off of protections-29;
- variation of the frequency of the electrical energetic system-8;

Main root human-caused initiators for accident situations:

-bad assembling-13;
-incorrect opened or closed valves-31;
-bad marked contacts or incorrect connected contacts-11;
-error by switch on/off of a equipment instead of another from the same type -11.

When such events occur on the higher level of the accident sequence, they influence the situation to the worse end state.
In the 27% from all protocol accident situations there are present human-caused errors or failures. This percent rate is near to the percent rate, calculated for the failures of the equipments of the safety related systems we spoke about. We can make the conclusion, that this value of the human-caused errors is typical for NPP "Kozloduy"- around 20%.There is here a big potential for decreasing of the number of failure and accidents occurrences. But here are included also bad maintenance and repairing, bad assembling, bad project and design performance, that increases this percent rate.

The failure rate λ for accident situation occurrence because of human-caused error or failure is assessed as

λ=5.0.10-4, 1/h.

Common-cause initiators for accident situations.

- loss of power supply for equipments or systems;
- loss of service air;
- loss of greasing water;
- loss of service water;
- loss of onshore pump station;

In the report [10] is researched the influence of the given common-cause initiators to the frequency of the occurrences of accident situations in the NPP "Kozloduy".

The failure rate for accident/incident occurrence during the years from 1974 till 1993 varies without any trend around the mean value of λ, assessed for this period for units 1- 4:

λ=1.5.10-3, 1/h. Increasing of the λ value is observed one or two years after a start of a new unit to work, because of a normal process of synchronising of the systems and the personnel in the new unit.

Accidents/incidents overview of the different units.

There are not big differences between the value of a failure rate for accidents λ for the different units 1-4. These values are in the interval

$\lambda \in$ (5.0.10-4--6.8.10-4), 1/h.

The most high value λ=6.85.10-4, 1/h , is calculated for unit 3 of the NPP "Kozloduy".

Accidents characteristics, because of mechanical causes.

λ=7.47.10-4, 1/h;

The mechanical caused events are 36% from the number of all described in the accident protocols events on NPP "Kozloduy".

Electrical caused accident events.

λ=8.31.10-4, 1/h;

The electrical caused accident events are 60.8% from all protocol events.

We can see that the two time greater percent rate of the electrical caused accidents, compared to the mechanical caused, has no influence to the failure rate's relation. The both values of λ are similar. This fact does not contradict the theoretical laws and we recommend the λ values as more representative.

As we wrote, the relation of the failures of the equipments because of electrical:mechanical causes varies between 1:20 for the emergency feed water pumps till 1:1 for the diesel generators and these relations are not valid for the reliability characteristics.

Accidents occurrences because of error signal . Reliability characteristics.

λ=3.7.10-4, 1/h. This is a relative big value and it could be strong reduced with measurements concerning gut testing and regulation of equipments or elements with protection functions.

Generally we can make the conclusion, that in the more of a half accident situations were present human-caused and/or error signal caused failures in the sequence of the accident process. Here we see a great area for measurements for reducing the accident/incident events in the NPP "Kozloduy" with not much investigations.

For all 59 initiating events (IE) from the list of IE [10] were calculated reliability characteristics and frequencies:

- failure rate for each accident/incident from the IE list: λ, 1/h;
- expected period without accident events T=1/λ, h.;
- percent rate of the occurrences for every accident/incident from the IE list, %;
- mean period without occurrences for every accident/incident from the IE list, h.;
- relative number of occurrences for every IE , 1/reactor-year.

The reliability characteristics λ, calculated for every initiating event on the basis of all protocols (1974-1993), are with small difference and are in the interval

$\lambda \in$ (3.0.10-5--6.0.10-4),1/h.

The most possible accident event is turbine trip with λ=6.2.10-4, 1/h;

Events with the least possible occurrence are inadvertent fell down of control rods and loss of power supply DC with λ=3.7.10-5, 1/h. These characteristics are calculated only for IE, occurred more often than 4 times during the period 1974-1993.

Assessment of some reliability characteristics for rarely events.

There are different methods for assessment of the reliability characteristics of events with small frequency. These methods include expert opinion, event and fault tree analyses and other indirect methods. In our analyses, based on exploitation data, we have calculated

the quantitative characteristic: number of initiating events/reactor-year, 1/reactor-year, for events with small frequency of occurrence (1-4 times). The value of this characteristic is between 0.0167 and 0.0668, 1/reactor-year. For events for which λ is calculated, we recommend this characteristic instead of the relative frequencies as more representative. Our assessment of the reliability characteristics were compared with similar data, published in [1]-an IAEA document-where are given reliability characteristics for some similar or identical IE for NPP with reactors WWER or PWR. The reliability characteristics, assessed for our reactors, are most similar to the characteristics from NPP "Loviisa"[1] and practically identical to these for NPP "Kozloduy", given in [1]. In [10] we commented some big differences in the degree of the characteristics by some of the ten IE, given in [1], comparing to the characteristics in our assessment [10].
Because of the full volume of data about the accident situations we assessed, overviewed and analysed, and after the performed check of the results and of the conclusions on different methods and ways, we can assert, that the reliability characteristics, calculated in our reports, are very correct.

Characteristics of events with more heavy consequences. Shut down of a reactor. Quantitative and qualitative characteristics for some relationships between some initiating events.

Shut downs of a reactor on the different units.

The shut down failure rate for the reactors of units 1-4 , NPP "Kozloduy" are between (1.14.10-4–3.08.10-4), 1/h. The worst value of is calculated for unit 3 and the best value - for unit 1. As we said, also by the analyses of the reliability of the equipments from some safety related systems, we obtained worse characteristics for the equipments from unit 3[9].
Great frequency can be seen by the accident's sequences with end state shut down of a reactor, caused from the followed direct causes:
-error signals from ionisation chambers;
-by switch-on of redundant power supply often the start voltage of sections 6kV is only 4kV and some times occurs loss of main coolant pumps (for example). By loss of 3 or more main coolant pumps follows shut down of the reactor.
λ=2.4.10-4, 1/h.- falure rate for shut down of any reactor from units 1-4 because of an occurred error signal.
λ=6.0.10-5,1/h.- failure rate for shut down of any reactor from units 1-4 because of direct error signal for shut down.
λ=1.02.10-4, 1/h.-failure rate for shut down of any reactor from units 1-4 because of variations of the electrical parameters of the electrical energy system.
λ=3.57.10-4, 1/h.failure rate for shut down of any reactor from units 1-4 because of turbine trip of last working turbine.
λ=4.3.10-5, 1/h.-failure rate for shut down of any reactor from units 1-4 because of loss of external power supply.
λ=1.28.10-4, 1/h-failure rate for shut down of any reactor from units 1-4 because of human-error.
λ=4.8.10-4, 1/h.-failure rate for shut down of any reactor from units 1-4 because of loss of 3 or more main coolant pumps.
λ=1.3.10-4, 1/h.-failure rate for shut down of any reactor from units 1-4 because of generator trip.
λ=8.2.10-5, 1/h.failure rate for shut down of any reactor from units 1-4 because of unpossibility of the regulator for the level in a steam generator ВП-12 to keep the level by big transients. This process usually goes through turbine trip till shut down of a reactor. Our conclusion and also the opinion of the personnel in NPP "Kozloduy"are, that the regulators for the level in the steam generators can't keep the level by big transients.

λ=6.5.10-5, 1/h.-turbine trip because of overfeed or low level in a steam generator;
λ=6.6.10-5, 1/h.-loss of both external and internal power supply AC at the same time.
There occurred one case with full loss of normal and redundant power supply AC and DC at the same time, because the Principe for divide the sources of power supply for the different redundant systems was not kept. This case was finished quickly with switch-on of an external power supply.

λ=3.3.10-5, 1/h.-radioactive releases or detected radioactivity. During the period 1974-1993 was not any accident or incident with radioactive releases outsides of the plant.

4. Conclusion.

In this paper we gave some data and characteristics about the failures of equipments of some safety related systems and conclusions, based on the reliability analyses of these equipments. Also we gave some qualitative and quantitative reliability characteristics, involved from the analyses of the accident protocols 1974-1993. If we compare the both types of analyses, we can see similarities, but also some incompatible sides of the both types of analyses. The accident situation includes complex sequence of failures or loss of functionality of equipments or systems because of dependent failures or protections switch off/on by big transients of the parameters. The most important differences are:

- the failures by demand by accident situations often are caused not from defect of an equipment but from loss of functions of other equipments or from common-cause initiators, safety or protection signals from relays and so on.
- in the accident protocols are not included all failures of the equipments, because the most number of failures haven't caused accident process and were not a part from such process.

But they exist also similarities and we can give some recommendations, based on the both type of analyses:

- there is necessary to be stooped the ageing and wearing processes with on-time changing or repairing of the equipments. The new elements must be as defined in the project standards;
- optimisation of the maintenance;
- periodical tests and inspection of the equipments, that normal are in stand-by state or are redundant, for on-time detection of defects with the goal to prevent accident processes;
- the Principe for dividing of the logical and physical schemes of different supply, safety or protection elements must be kept;
- bettering and increasing the level of competence of the personnel of the NPP;
- reduction of the common-cause initiators for failures;
- modernisation of the element base and testing the new decisions from the point of view of reliability and compatibility with the system as unit;

There exist also similarities in the quantitative and qualitative characteristics in the both types of analyses:

- type of the root cause:
  - -electrical;
  - -mechanical;
  - -human-caused errors;

On the basis of the performed analyses were made useful conclusions and recommendations about measurements for increasing the safety and the ability of the equipments for better task performance.
After the overview of the full volume of the information in the accident protocols of NPP "Kozloduy"-units 1-4 we can make the conclusion, that in the NPP "Kozloduy" in this period there were not occurrences of accidents with consequences for the fuel integrity or radioactive releases outsides of the plant.

The percent of human-caused errors is relative big, more than 20%, but we can say, that the personnel of the plant works good and with competence as in more simple also in more complex situations, is gut educated and high professionally, that prevented till now occurrences of heavy accidents in the NPP "Kozloduy".
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With not much effort could be increased the reliability of the systems with reduction of the human-caused errors and of the occurrences of error signals.
Much more invest is necessary for eventually reconstruction and modernisation because of the needed variation reliability analyses, new projects, tests and proving the compatibility of the new projects with all other systems of the NPP "Kozloduy"

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ВЕРОЯТНОСТНЫЙ АНАЛИЗ БЕЗОПАСНОСТИ I - IV БЛОКОВ АЭС "КОЗЛОДУЙ" С РЕАКТОРАМИ ТИПА ВВЭР- 440 (В 230) ПРИ ВКЛЮЧЕНИИ НЕЗАВИСИМОЙ СИСТЕМЫ АВАРИЙНОЙ ПОДПИТКИ ПГ

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При определенном спектре аварийных режимов для приведения реактора в безопасное состояние необходимо выполнить функцию - "отвод остаточного энерговыделения" от активной зоны.

В проекте реакторов ВВЭР 440 (В230) выполнение этой функции в любых аварийных режимах осуществляется по второму контуру.

При определенных аварийных режимах, таких как - полная потеря питательной воды, разрыв питательного коллектора, инициированных отказами по общей причине в машзале (землятресение, затопление, пожар), невозможно осуществлять подпитку ПГ через существующие патрубки и для выполнения выше упомянутой функции необходимо использовать другие системы или процедуры.

Одна из таких возможностей - применение независимой системы аварийной подпитки ПГ (НСАПВ). Эта система обеспечивает выполнение функции отвода остаточного энерговыделения от активной зоны при следующих аварийных режимах:

- полная потеря питательной воды;
- разрыв коллектора питательной воды;
- разрыв трубки ПГ;
- полное обесточивание;
- разрыв парового коллектора.

Проведенные термогидравлические анализы этих аварий показывают, что лимитирующие событие, при котором рассчитывается система - полная потеря питательной воды [1].

Физическая природа протекающих процессов этой аварии заключается в том, что уровень в компенсаторе объема(КО) достигает максимальных значений в начале аварии и в следующих этапах ее развития, в результате разогрева первого контура, в процессе осушения ПГ. Особенности развития этой аварии в том, что первое открытие предохранительного клапана компенсатора объема (ПК КО) происходит на 640 s с начала транзиента, а позднее начинается циклическое открытие ПК КО, которое приводит к потере теплоносителя первого контура. В той же самый момент начинается разогрев первого контура из за уменьшения эффективности теплопередачи через трубной пучок ПГ. Это является и началом выключения независимой системы аварийной подпитки ПГ с расходом питательной воды - 65 m^3/h.

В [1] подробно описаны ход развития аварии и обоснование структуры системы. На рис. 1 представлена принципиальная схема НСАПВ.

Структура системы двухканальная (с учетом приницила единичного отказа), каждый канал включает насос, арматура, система эл. питания и система КИП и А. Параметры каждого насоса таковы - расход 65 m^3 /h, напор 4.8 MPa. Два бака с химобезсоленной водой подпитывают систему. Каждый канал системы питается собственным дизельгенератором. Для каждого бака система размещается в отдельном здании (сеизмоустойчивое), вне реакторного отделения и машинного зала. Система включается оператором ручным способом. Для оценки роли НСАПВ, в улучшении безопасности блоков, проведен вероятностный анализ безопасности с применением метода "Дерево событий" (ДС). С помощью этого метода моделируется поведение блока при определенном исходном событии и оценивается частота последствий для активной зоны.

Вероятностные анализы проведены для I - IV блоков АЭС "Козлодуй" для двух случаев - при включении существующих систем безопасности блоков и при включении НСАПВ. Анализы проведены для исходного события "полной потери питательной воды".

При этом сценарии развития аварии предполагается, что полная потеря питательной воды и аварийной подпитки ПГ инициируется отказами по общей причине в машзале. Обесточивание блока наступает в момент генерирования сигнала АЗ I рода по низкому уровню в 2 из 6 ПГ.

Время важнейших событий протекания аварии представлены в табл. 1.

Таблица 1
Развитие аварийного процесса

No	Событие	Время, сек
1.	Полная потеря питательной воды	0.0
2.	Срабатывание аварийных стопорных клапанов турбины	45.0
3.	Срабатывание АЗ I рода	48.0
4.	Обесточивание блока	48.0
5.	Конец электромеханического выбега ГЦН	228.0
6.	Первое открытие ПК КО	640.0
7.	Прекращение эффективного отвода тепла через ПГ	11.400
8.	Заполнение КО	12 240
9.	Достижение состояния насыщения в первом контуре	14 300

Моделирование поведения блоков и расчет деревья событий осуществляется при следующих исходных условиях и допущениях:

1. Предполагается, что после срабатывания СУЗ, реактор приводится в подкритическое состояние.

2. Последствия для активной зоны при несрабатывании СУЗ (ATWS) и при полном обесточивании блока без срабатываний ДГ не рассматриваются [3].

3. Частота исходного события принимается - 2.10^{-3} 1/РГ [2].

4. Критерий успеха независимой системы аварийной подпитки ПГ - работа одного канала системы, обеспечивающей расход питательной воды в ПГ 65 m^3/h.

5. Данные неготовности систем безопасности, участвующие в овладевание аварии, даны в табл. 2.

6. Для систем безопасности I и II блоков АЭС "Козлодуй" учтены проведенные до сих пор изменений.

7. Результаты расчета неготовности НСАПВ представлены в [1].

Таблица 2
Данные неготовности систем безопасности I - IV блоков АЭС "Козлодуй"

No	Система безопасности	Неготовность систем безопасности I и II блоков	Неготовность систем безопасности III и IV блоков
1.	СУЗ	2.4×10^{-5}	2.4×10^{-5}
2.	Система аварийного электропитания	2.56×10^{-3}	1.69×10^{-3}
3.	Система аварийного подпитки ПГ	5.22×10^{-3}	1.13×10^{-4}
4.	НСАПВ	3.83×10^{-3}	3.83×10^{-3}
5.	Система аварийной подпитки первого контура	1.9×10^{-3}	2.99×10^{-4}
6.	Спринклерная система	3.71×10^{-3}	3.68×10^{-4}
7.	Отказ на открытие ПК КО	4×10^{-3} 1/D	4×10^{-3} 1/D
8.	Отказ на закрытие ПК КО	7.0×10^{-3} 1/D	7.0×10^{-3} 1/D

На рис. 2 и 3 представлены деревья событий для двух анализированных случаев.

В табл. 3 представленные полученные результаты расчета частоты последствий для активной зоны I - IV блоков в АЭС "Козлодуй".

Анализ полученных результатов показывают, что частота повреждения активной зоны для аварии с включением НСАПВ ниже по сравнению со случаем, когда включаются только существующие системы.

Из этого следует, что при применеии НСАПВ улучшается безопасность I - IV блоков АЭС "Козлодуй".

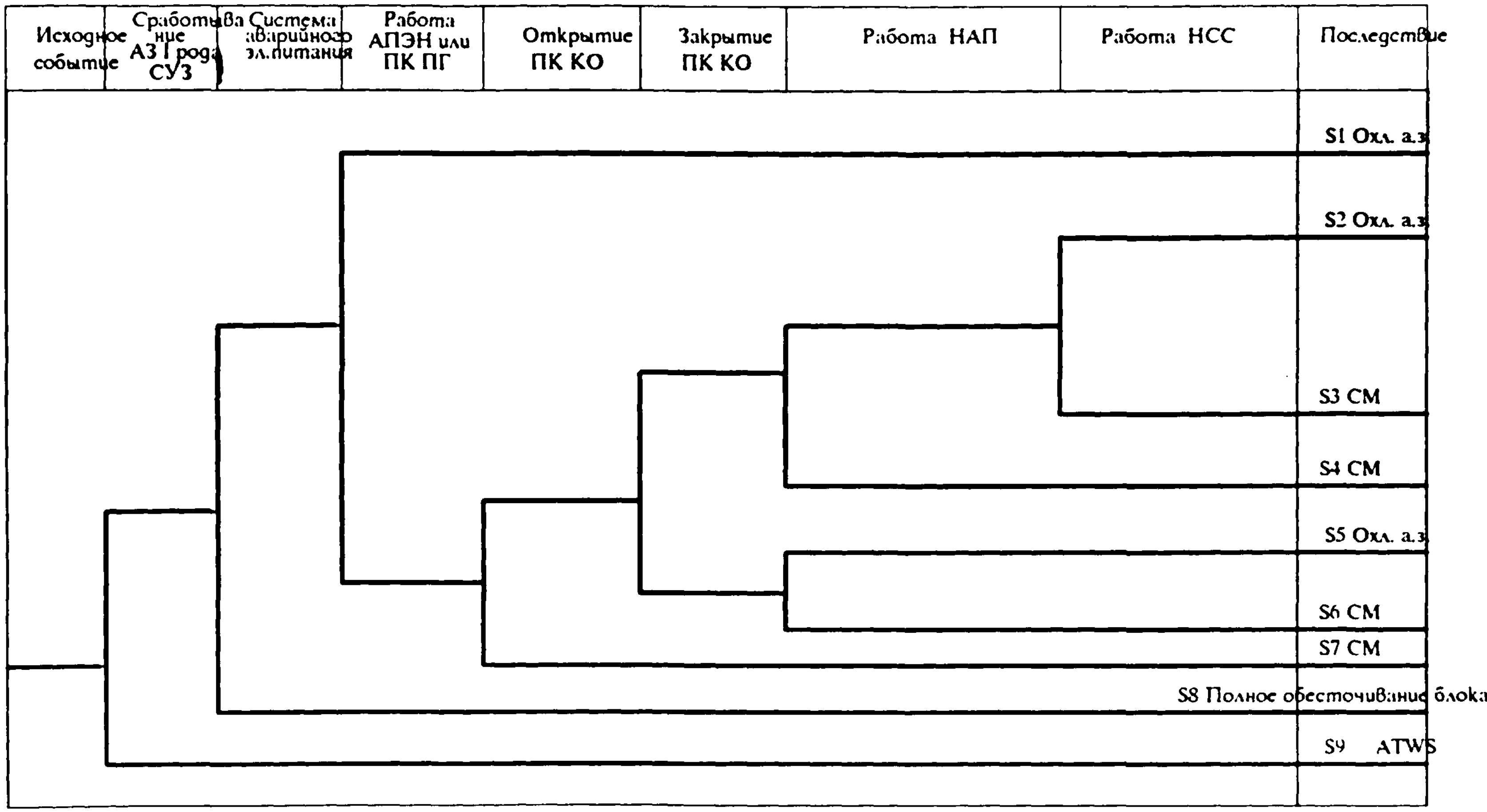

Рис.2. Дерево событий аварии "Полной потери питательной воды" при включении сoществующих систем безопасности блоков.

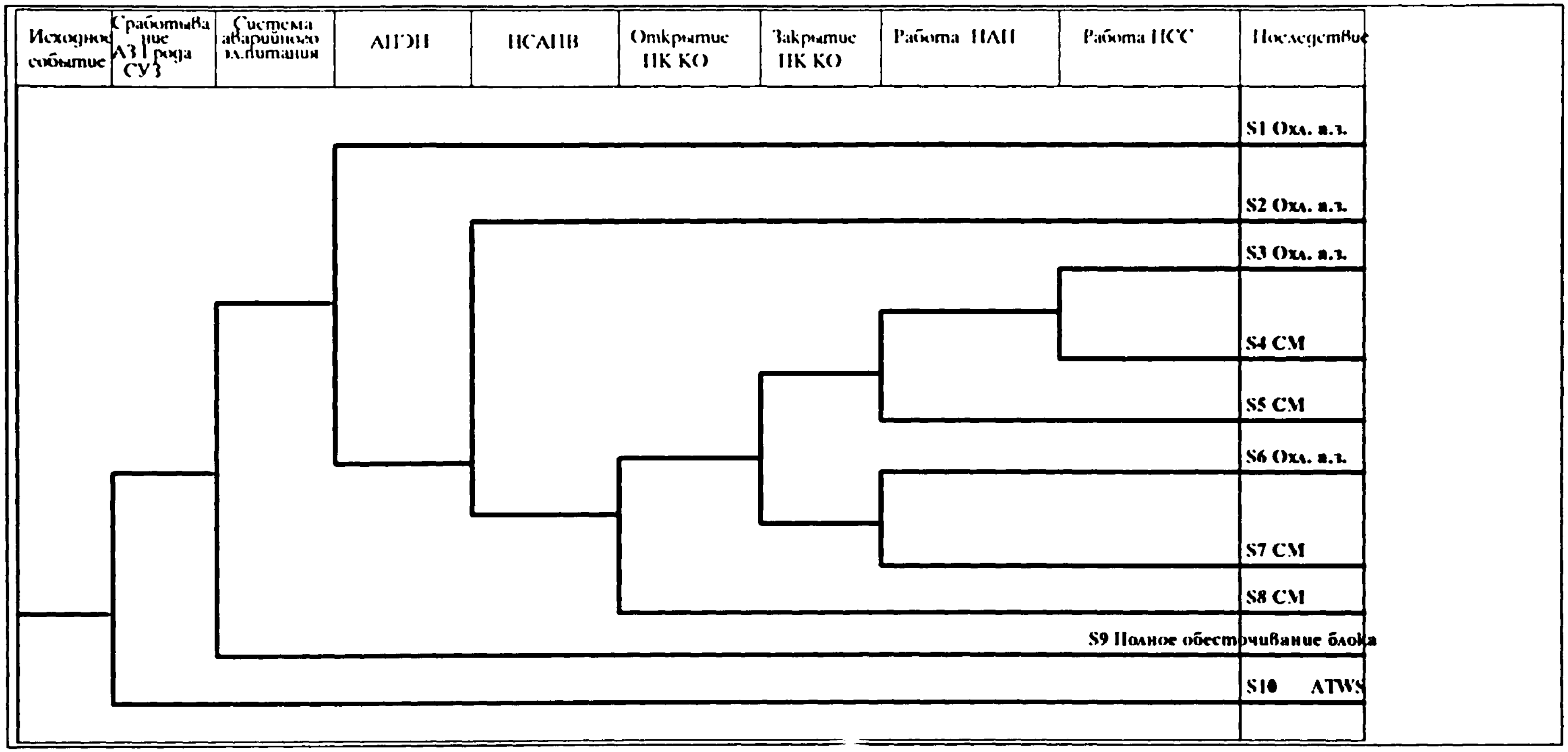

Рис.3. Дерево событий аварии "Полной потери питательной воды" при включении независимой системы аварийной подпитки ПГ.

Таблица 3

Результаты расчета последствий для активной зоны I - IV блоков ЭС "Козлодуй" при включении существующих систем безопасности и при включении НСАПВ

No	Последствие	Частота последствий для аварии "полной потери питательной воды" при включении существующих систем безопасности		Частота последствий аварии "полной потери питательной воды" при включении НСАПВ	
		Бл. I и II	Бл. III и IV	Бл.I и II	Бл.III и IV
1	S1	1,98 10^{-3}	1,99 10^{-3}	1,98 10^{-3}	1,99 10^{-3}
2	S2	6,32 10^{-6}	2,22 10^{-7}	6,39 10^{-6}	2,24 10^{-10}
3	S3	3,84 10^{-8}	8,19 10^{-11}	2,27 10^{-8}	8,43 10^{-10}
4	S4	6.64.10^{-8}	6,68 10^{-11}	8,96 10^{-11}	3,14 10^{-12}
5	S5	7,25 10^{-8}	1,56 10^{-9}	1,55 10^{-10}	5,41 10^{-12}
6	S6	4,60 10^{-10}	4,69 10^{-12}	1,7 10^{-10}	5,96 10^{-10}
7	S7	4,26 10^{-8}	9 10^{-10}	1,49 10^{-12}	3,82 10^{-12}
8	S8	Полное обесточивание	Полное обесточивание	9,84 10^{-11}	3,44 10^{-12}
9	S9	ATWS	ATWS	Полное обесточивание	Полное обесточивание
10				ATWS	ATWS
11	Суммарная частота повреждения активной зоны	1,48 10^{-7}	1,08 10^{-9}	3,44 10^{-10}	1,2 10^{-11}

Выводы

Проведенные анализы безопасности I - IV блоков АЭС "Козлодуй" показывают, что для выполнения функции отвода остаточного энерговыделения при исходном событии "полная потеря питательной воды", (инициированного отказами по общей причине в машзале), необходимо включить независимую систему аварийной подпитки ПГ.

Вероятностные анализы безопасности показывают, что при включении этой независимой системы повышается безопасность I - IV блоков АЭС "Козлодуй", которая выражается в снижение частоты повреждения зоны при рассматриваемом исходном событии.

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MODELLING AND SIMULATION OF PROCESS CONTROL SYSTEMS FOR VVER

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A new method is developed for modelling of the dynamic properties of large class of objects in power industry. A high accuracy is observed and ability for real time simulation of process control systems.

A short description of the main advantages of the method for modelling of controlled systems for NPP and some applications are given.

The method is developed to satisfy the specific needs of the control for VVER units NPP "KOZLODUY" as follows:

To give opportunities for analyses and design of control systems.

To give opportunities for process control system simulation.

To be a basis for estimation of immeasurable parameters important to safety.

To be a basis for simulation of different systems of all units with aim of malfunction diagnostic, prediction of dynamic behaviour, and control vector estimation.

I. DYNAMIC BEHAVIOUR MODELLING OF SYSTEMS FOR VVER

For many objects the models based on known physical relationships are unsatisfactory and very complicated from modelling and mathematical point of view. They need allot of efforts to construct an adequate model. The numerical methods and speed of calculations often don't allow a real time simulation of complicated objects and their optimal control, diagnostic and prediction of dynamic behaviour.

With the purpose to avoid these disadvantages and specifically for the investigation of control systems a new modelling technique is developed(a method for identification) - based on least scares and treating highly efficient linear uninterrupted equations.

It is well known that differential equations of type

$$\sum_{k=0}^{n} a_k y^{(k)} = \sum_{k=0}^{m} b_k x^{(k)}$$

in certain conditions are considered as models of dynamic behaviour of large class uninterrupted objects, used in power industry, chemistry, electrical engineering etc.

The coefficients a_k and b_k can be obtained after linearization of preliminary known models. However, the most precise information could be obtained, using input/output signals. Based on these signals, the new modelling technique can provide mathematical model which "held" information of the dynamic features of the objects.

A short description of the capability of the new method, the basic features of which were developed in [3], will illustrate the software system's capability for identification of orders "n" and "m" and the coefficients a_k and b_k.

Different methods are known, depending on using filter of signals. They are created with the purpose to avoid the problems connected with accurate estimation of parameters and presence of noise.

The suggested method has the advantages as follows:

- There are no significant limitations on the type of input/output signals

- There are no significant limitations on the length of data time series.
- The identification is used at none zero initial conditions which concede with the real process

A high accuracy is observed in case of noise.

The program system uses and realises a number of original algorithms and procedures, including;

- An algorithm for automated processing of experimental time series;
- An algorithm for passing from a differential equation to an integral equation;
- An algorithm for multiple integration that doesn't introduce additional error;
- Method for calculating transient motions that allows solution of differential equations with presence of a differential polynomial in the right side, presence of noise and jumps of input signals, presence of transport delay (dead time)
- Optimising the order and coefficients of differential equation procedure
- In addition a new efficient procedure is developed for dynamic filtering of signals, using the method for calculating of transient motions.

1.1. TEST EXAMPLES FOR IDENTIFICATION BY INFORMATION OF INPUT/OUTPUT SIGNALS.

The output of preliminary specified model (transfer function) is calculated by information of input signal. The purpose of the testing is to guess the parameters of the model (to restore the transfer function) by information of input and output. These are test examples and are not related with certain objects.

On fig.1 and fig.2 input and output signals are shown. The output is calculated by using 100 discreet values of input. The specified structure of the model is as follow:

$$\boxed{00y^{(2)} + 10y^{(1)} + y^{(0)} = x^{(0)}}$$

The computer calculation results (results of identification) are as follows:

- The identification procedure are started by using 10 points of smooth interval from 51 to 61 point.

$$00.000000796y^{(2)} + 9.9999999552y^{(1)} + y^{(0)} = 1.0000000004x^{(0)}$$

- The input and output are transformed as it shown on fig.3 and fig.4. The identification used 25 points of noisy interval from point 10 to point 35.

$$80.58226y^{(3)} + 94.23973y^{(2)} + 8.55884y^{(1)} + y^{(0)} = 0.96143x^{(0)}$$

- The identification used 50 points of noisy interval.

$$99.72799y^{(2)} + 8.75733y^{(1)} + y^{(0)} = 0.99082x^{(0)}$$

If are used 50 discreet values from time 0 to time 5 the result is as follow:

$$.333y^{(4)} + 0.267y^{(3)} + 100.000y^{(2)} + 10.000y^{(1)} + y^{(0)} = 1.000x^{(0)}$$

On fig.5 and fig 6 input and output signals are shown. The output is calculated using 150 discreet values of input. The specified structure of the model is as follow:

$$\boxed{00y^{(2)} + 10y^{(1)} + y^{(0)} = 50x^{(2)} + 2x^{(1)} + x^{(0)}}$$

Identification used first 100 discreet values of noisy interval

$$03y^{(2)} + 9.56244y^{(1)} + y^{(0)} = 50.23119x^{(2)} + 1.72202x^{(1)} + 0.97137x^{(0)}$$

-Identification in presence of jumps. Interval from point 101 to 150 is used

$$04.49929y^{(2)} + 11.67857y^{(1)} + y^{(0)} = 52.65466x^{(2)} + 2.57956x^{(1)} + 1.03539x^{(0)}$$

1.2. DYNAMIC BEHAVIOUR MODELLING FOR STEAM GENERATOR LEVEL.

The steam generator level control is of great importance for safety operation of the units. The increased number of failures, which in certain cases lead to reactor trip, supposed incorrect behaviour of level controllers and imperfect control design.

. The presence of (suitable)appropriate models for dynamic of basic channels connected with level control in steam generator, give opportunity for controllers' parameters optimising at different level of power and different structures of control system investigating.

Fig.7 shows identification of real object - input signal - steam flow to turbine and fig.8 shows output signal - SG level and model calculation presented by the dots.

The model has following form:

$$0.19769y^{(2)} + y^{(1)} = 0.70915x^{(1)} - 0.70905x^{(0)}$$

In this case for simulation purpose the desired input must be specified in minutes, or the model must be transferred by simple multiplying of the coefficients.

With the help of restored transfer functions of channels:

- steam flow -level
- feed wåter flow - level
- control rods position.- level
- MCP disturbances - level

and also taking into account transfer functions of detecting devices, the controller and flow rate characteristic of main control valve, becomes easy to obtain the next transfer function, describing level dynamics.

$$H = F(p)^* Xu + Y1(p)^* Gn + Y2(p)^*Q$$

were H is level in steam generator; Gn - steam flow ;Q - heath from primary circuit

F1(p) - The transfer function of the closed system of level by manipulated signal

Y1(p) - The transfer function of the closed system of level by disturbance -steam flow.

Y2(p) - The transfer function of the closed system of level by heat transferred from primary circuit

By analogy it's easy to obtain the transfer function, describing the feed water flow dynamic behaviour caused by steam flow, heat from primary circuit and manipulated signal.

For VVER - 1000 it's very useful to obtain the transfer functions of feed water flow and steam flow to turbine(to estimate on-line their dynamiv behaviour), because of imperfect control design and instrumentation(lack of steam flow signal, low reliability).

II. PROCESS CONTROL SYSTEM SIMULATOR BASED ON KASKAD-2 FOR VVER-1000

The Process Control System Simulator(PCSS) was created for the training purposes of operating personnel of VVER-1000 and it uses models created by the described method It is needed generally for the following reasons.

-Lack of simulators and special training tools at Kozloduy NPP at all.

-The increased number of control systems failures

-Lack of information for control systems behaviour and the characteristics of the objects of the control

-Opportunity to use the simulator for various investigations(researches) of the control systems and hardware reliability.

PCSS simulates different control loops, based on KASKAD Control System(modules and controllers)

The simulator consists of two main units.

-KASKAD Control System, which is replica of the existing VVER-1000 Control System. The control units are placed in the cabinets, which are replica of the cabinets, used at the VVER-1000 units.

-Mathematical model of the process. There is an option that allows to change the models according to the trainer's desire. The mathematical models are implemented on a personal computer with a CRT high resolution. This way the trainees will be able to follow the process development.

-The interface between the object of the control(mathematical model) and the Control System was designed specially for the simulator. The speed of data transfer between the computer and the Control System is corresponding to the real speed of the signal in the existing systems.

During the training procedures the trainer will be able to set up different configuration of the KASKAD-2 Control System and simultaneously to choose the preset adjustment of the controllers. Using the menu facilities, the trainer will be able to set the model of the object, choosing one of the following options:

-model as transfer function.

-model as an element - for instance instanetiones element, integrator, differentiator, etc.

-model as input data files from experiment. Using these data the computer can calculate the mathematical model.

The trainees can observe on CRT the system transient and vary the adjustment of the controllers in order to improve the transient performances. This way they can see how the configuration of the Control System and/or the new tuning can affect the quality of the control, i.e. the degree of the transient damping.

The transient is calculated by the described method for direct solving of differential equation of high order. The method involves a preliminary reduction of the differential equation to an integral equation and taking into account the transport delay(dead time) of the object [4].

The simulator offers good opportunities for training and investigating the transients and the control systems as far as full replica of KASKAD is used.

The resulting advantages are as follows:

-operational personnel at the NPP improvement of training quality.

-opportunity for hardware testing of KASKAD units

-improvement of control systems operation and optimising of control systems parameters.

-different structure of the control systems examination.

III. CONCLUSIONS

The developed method shows high accuracy and reliability for modelling and simulation of process control systems.

There are no significant limitations on the type of signals

Identification at none zero initial condition is possible.

On line identification is possible.

On the basic of real experiments and simulated with known computer codes data time series it is possible to construct highly efficient models of different systems or all units for solving the problems as follow:

- Real time simulation with high accuracy for training purposes
- Estimation of immeasurable parameters important to safety.
- Malfunction diagnostic on the basis of plant dynamics.
- Prediction of dynamic behaviour.
- Control vector estimation in regime adviser.

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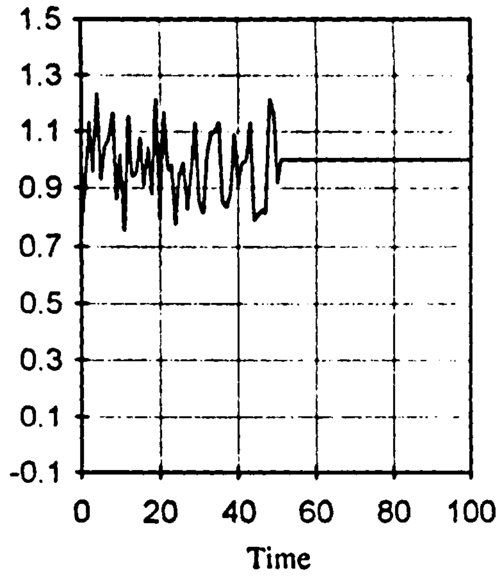

Fig.1. Input signal

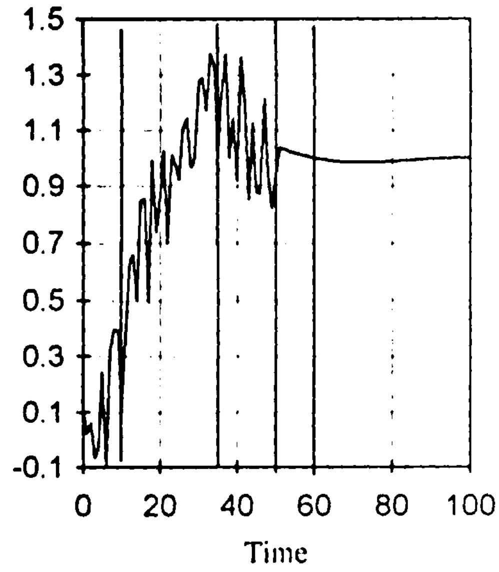

Fig.2. Output signal

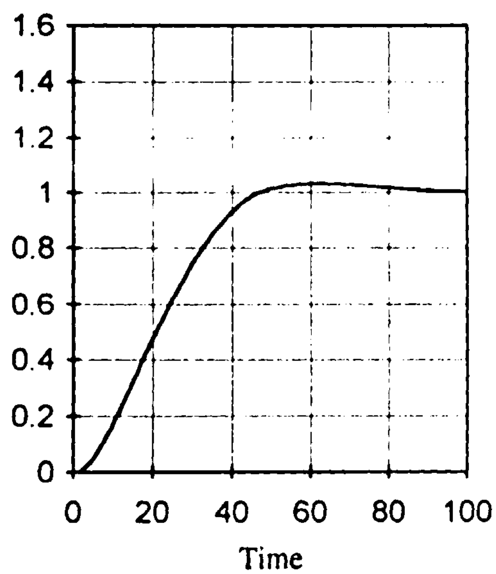

Fig.3. Input signal transformed

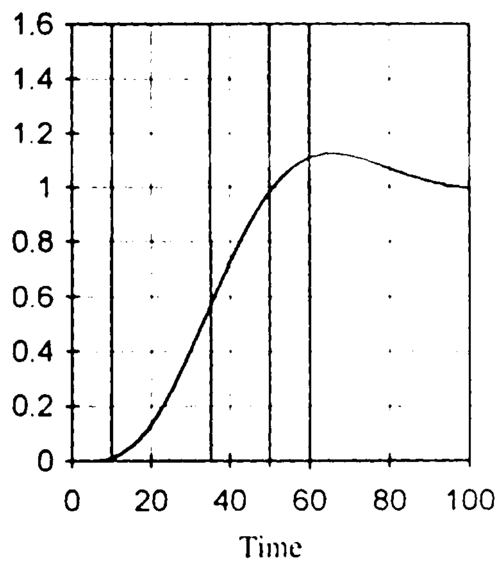

Fig.4. Output signal transformed

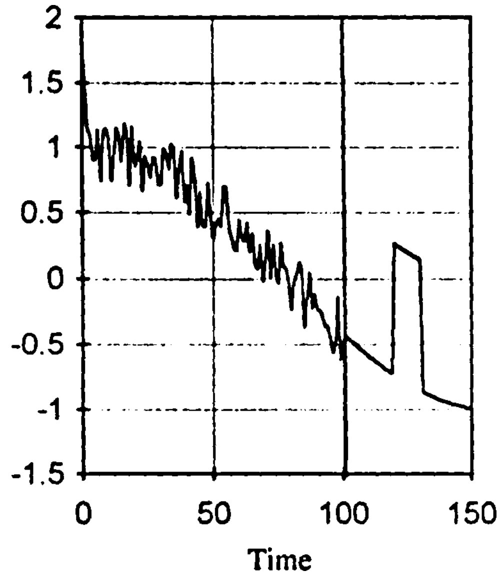

Fig.5. Input signal

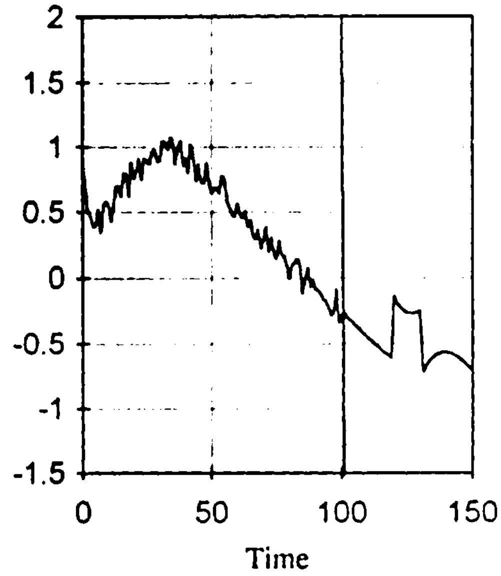

Fig.6. Output signal

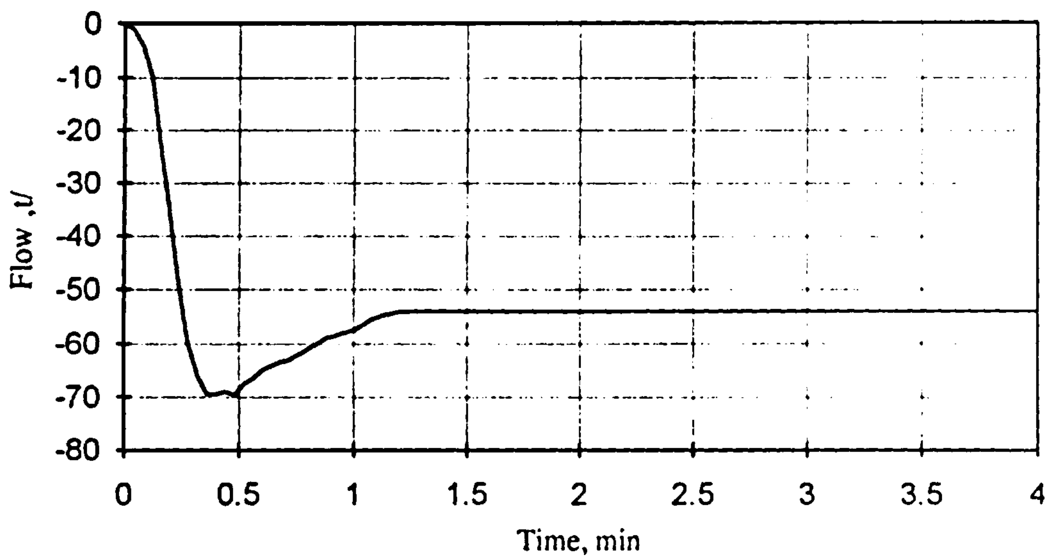

Fig.7. Steam flow to turbine

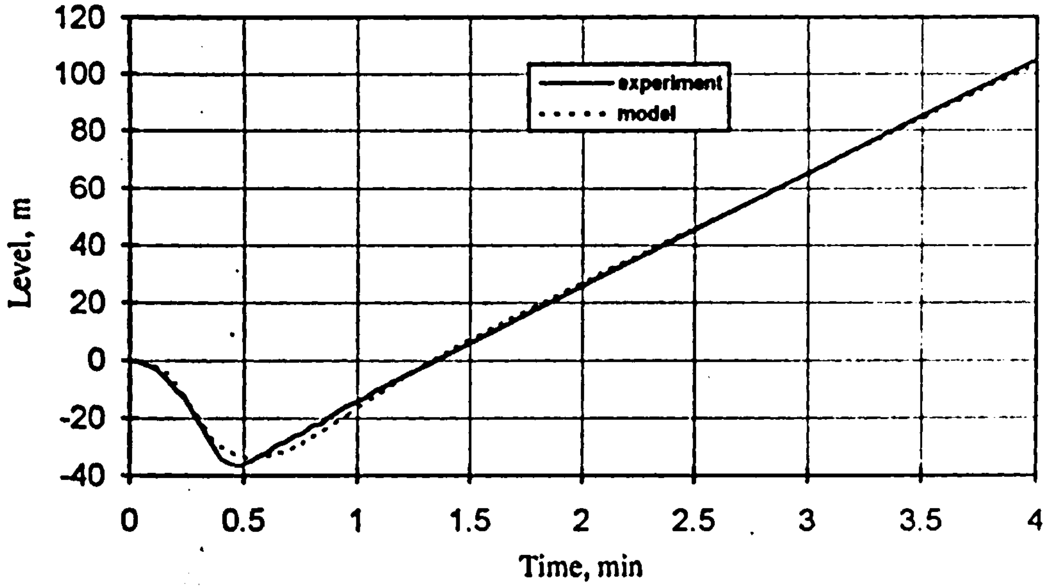

Fig.8. SG level

BG9600413

Human Reliability Analysis as part of Probabilistic Safety Assessment - Level 1 for Kozloduy - 3 NPP

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Risk Engineering Ltd.

1. Scope of the Study
2. Quantitative Methods for HRA
3. Modelling Assumptions
4. Human Error Probability Quantification Methodology
5. Example

1. Scope of the study

The purpose of the Human Reliability Analysis is to identify and quantify the Human errors that could significantly affect the frequencies of accident sequences evaluated in this study.

The human errors considered in this study are of two distinctly different kinds[3]:

1. Errors made during normal plant operation, before the initiation of any accident sequence, so called "preaccident" actions;
2. Errors made in responding to an accident sequence - so called "postaccident" actions.

There is no consensus on the best methods to perform HRA. Limitations in the applications of various methods include the following:

1. Human behaviour is a complex subject that does not lend itself to a simple models like those used for component and system reliability. Therefore HRA is more dependent on the judgement of the analysts.
2. Human actions are not only with binary success or failure logic as equipment failures.
3. There is general lack of data on human behaviour and the available generic data may not be applicable in different countries.

2. Quantitative methods for HRA

In the early 80's was published information on several methods for quantitative estimates of human performance [5]:

1. Fast simulation models (MAPPS, SAINT, etc.);
2. Expert judgement Methods (SLIM, STAHR, etc.);
3. Analytical Methods:

- time dependent activities - THERP, HCR, ASEP, OATS, etc.
- time independent activities.

Included in the category of "preaccident" errors are errors it testing, maintenance and calibration. For analysing and quantifying errors of the second type, the team used two recommended references:

1. EPRI-NP-3583 Systematic Human Action Reliability Procedure SHARP, Hannaman and Spurgin, 1984

2. NUREG/CR-4772 Accident Sequence Evaluation Program ASEP, A. P. Swain, 1987

ASEP is based on THERP method, which is very detailed and uses an extensive task analyses for any human action that is evaluated. It concentrates on mechanical tasks with little analysis on the thinking before action. But in most cases the operator's failure to identify the correct goal will dominate over errors at the level of individual action.

The team was advised to use a part of the modified SLIM method for qualitative assessment of HER, taking into account a limited number of performance shaping factors (PFSs) [5]:

- indications and alarms;
- complexity of the action;
- preceding action influence;
- response time;
- procedures;
- training and experience;
- stress.

Due to lack of time to provide interviews with plant personnel and to process these data the team decide to perform only general "screening" analysis.

3. Modelling Assumptions

A. Modelling of "repair" actions.

These include all dynamic human actions that are evaluated in the PSA models to prevent core damage or offsite release after an initiating event occurs. "Recovery actions" may involve such action as realignment of available equipment to replace failed components, use of an alternative system to maintain core heat removal (e.g. feed and bleed cooling) or manual operation of equipment after failure of automatic actuation signals. "Repair action" involve restoration of failed equipment during the accident event sequence. There are many factors that influence the effectiveness of possible equipment repairs: for example specific component type and failure mode, the amount of time that is available for repair, the type of accident, the location of failed equipment, etc. These factors cannot be evaluated realistically until information is available about the entire sequence of events. It was recommended that no "recovery" actions be included in any of the fault treed or event trees.

B. **Operator actions to back up failed automatic signals.**

It was recommended that the system fault tree models should not include operator actions to manually back up failed automatic signals. The PSA should evaluate these potential recovery actions only if the automatic signal failures are important contributors to core damage. It is very difficult process to evaluate operator's contribution for manual back-up. On the other side experience from many completed PSA has shown that automatic signal failures are usually very small contributors to system unavailability, even when manual back up actions are not considered. Therefore it is much more efficient and more correct to omit these actions from the fault trees.

C. **Combination of human errors and hardware failures.**

During system modelling for PSA Level 1 for Kozloduy - 3 NPP the team took into account the main recommendations of IAEA Review Mission all dynamic human actions to be included as separate top events in event trees instead if basic events in the fault trees. This practice clearly documents how these operator actions are included in the models and it displays important dependencies that affect operator's performance.

There is only one operator action that is included in the fault trees - it concerns RL51,52 - Main Feedwater System. It is "operator's failure to open MOV RL51,52S05. This system fault tree is used in transient event trees. It is clear to include this action in the fault tree because in case of operator failure the system RL51,52 will work through other MOV RL51,52S04. This action follows Emergency Instruction, it doesn't depend on time, it is "routine" action, which is done twice or three times per year, it is very simple and doesn't lead to emergency situation. The HER for this action is $1. 10^{-3}$.

It was also recommended that the human action and hardware failure should not be combined in a single top event (Fig 1.).

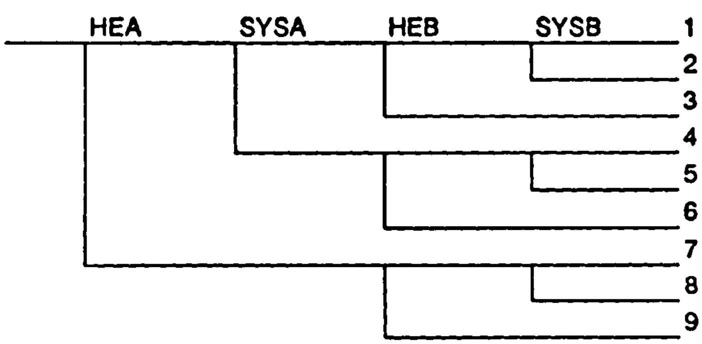

Fig. 1.

HEA - human action portion of top event A
SYSA - hardware failure portion of top event A
HEB - human action portion of top event B
SYSB - hardware failure portion of top event B

D. **Sequences of human errors.**

It is very difficult to assess dependencies between different human actions. Here was followed part of ASEP methodology for the second action, but it was modified in order to be more conservative, because of the lack or Emergency Operating Procedures and Training Simulator for Kozloduy NPP.

All subsequent dynamic human actions should be failed if the time between two actions is less than 30 minutes.

This "screening" model conservatively assumes complete dependence between first and all related actions that are required within 30 min. after the error. If the second is required after more than 30 minutes than it's HER is:

30÷60 min 0.25

>60 min 5.10^{-2}

E. **Top event "feed and bleed cooling".**

This top event is included in the model to evaluate operator actions to start feed and bleed cooling after loss of all steam generator heat removal. But there are many reasons to exclude this action from analysis. There are no thermal-hydraulic analyses to determine weather F&B cooling may be effective in Kozloduy NPP. The Kozloduy Operating Procedures do not include instructions for F&B cooling and the plant operators are not generally aware of this cooling option. Therefore is seem quite reasonable to take no credit for F&B in the "screening" quantification, where the team assumed that top event F&B is failed for all initiating events. This top event is kept in the event tree diagrams to document how F&B should be considered in the real PSA. The team performed second quantification to determine the importance of F&B cooling for reducing the Kozloduy core damage frequency.

4. Human error probability quantification procedure

There are many objective and subjective factors which became reasons to do only "screening" analysis. In order to reach stable global results the team should perform nominal analysis with plant specific data. By now due to lack enough knowledge, lack of experience in this field, lack of time and funds the team performed only this global "screening" analysis only on dynamic" human actions. So called "preaccident" human actions will be included in further modifications of this study.

The team used ASEP [3] to evaluate human errors. Table 1 is summarised information from fig. 7-1 and table 7-2 from [3]. The values are median and they should be transformed to mean values following the rules from table 20-20 from [2].

Table 1.
Human Error Probabilities
"Screening" values, based on NUREG/CR-4772

		Stress / difficulty of the action		
		High	Medium	Low
Time	Short	1.0	0.70	0.20
	Middle	1.0	0.10	1.0 10^{-2}
	Long	1.0 10^{-2}	1.0 10^{-3}	1.0 10^{-4}

5. Example - Small LOCA

The corresponding core damage frequency for this event 4.63E-4/y for the case with unavailable power supply. The sequences that contribute most of all to the core damage frequency are:

- Initiator - TQn2 (LPIS) failure - (1.8E-4/y) - sequence 2;
- Initiator - TXn0 (EFWS) failure - Operator failure H3YRDP2 - (1.5E-4/y) - sequence 6;

The unavailability of the systems included in the event tree sequences is caused mainly by DGs and electrical supply components failure. The first of the above mentioned sequences can be caused by the two DGs - GW and GX - failures. The reason for this is that in case of those both DGs failure there is no possibility for decay heat removal from the primary circuit via TQn2 system and recirculation. The operator failure is the biggest contributor to the core damage frequency for the second of the above mentioned sequences.

The full diagram of the event tree is shown on the fig. 2 :

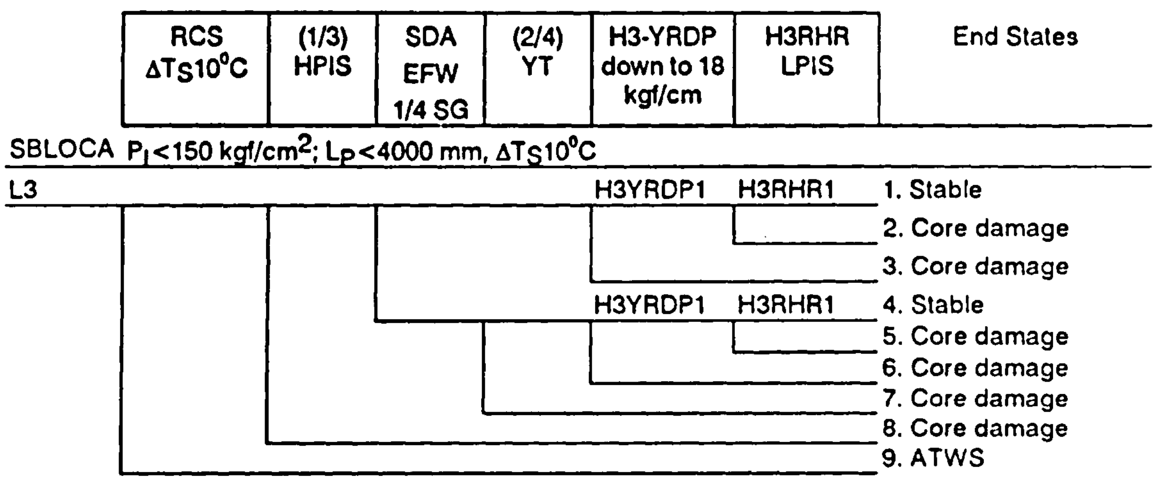

Fig. 2

Scenario Sheet for Human Action H3YRDP1

Human Action Name:	Operator cools down and depressurizes RCS
Human Action Identifier:	H3YRDP1
Event Tree:	SBLOCA
General Description:	The task is to decrease pressure down to 18 kgf/cm^2
Scenario:	1. Reactor scram actuates by: L_P < 4000 mm; P_I <150 kgf/cm^2; D t_S < 10°C/h. 2. High Pressure Injection System is available 3. Steam Dump to the Atmosphere are available
Task:	Operator should reduce RCS pressure below approximately 18 kgf/cm^2. This action is necessary to allow to accumulators to inject into the RCS before core subcooling is lost. Operator Should actuate switches on Safety Systems Panel in the Main Control Room (MCR). Meanwhile he must observe the level in the Hydroaccumulators (YT) and annunciation showing that the pressure has reached 18 kgf/cm^2.
Time Window:	>30 min after initiating event
Difficulty (stress)	H
Consequence of actions	first
HEP:	0.0161

Scenario Sheet for Human Action H3YRDP2

Human Action Name:	Operator cools down and depressurizes RCS
Human Action Identifier:	H3YRDP1
Event Tree:	SBLOCA
General Description:	The task is to decrease pressure down to 18 kgf/cm^2
Scenario:	1. Reactor scram actuates by: L_P < 4000 mm; P_I <150 kgf/cm^2; D t_S < 10°C/h. 2. High Pressure Injection System is available 3. Steam Dump to the Atmosphere are not available 4. Hydroaccumulators inject into the RCS.
Task:	Operator should reduce RCS pressure below approximately 18 kgf/cm^2. This action is necessary to allow to accumulators to inject into the RCS before core subcooling is lost. Operator Should actuate switches on Safety Systems Panel in the Main Control Room (MCR). Meanwhile he must observe the level in the Hydroaccumulators (YT) and annunciation showing that the pressure has reached 18 kgf/cm^2.
Time Window:	>10 min after initiating event (here the time is less because if Steam Dump to the Atmosphere is not available operator should decrease pressure earlier).
Difficulty (stress)	H
Consequence of actions	first
HEP:	1

Scenario Sheet for Human Action H3RHR1

Human Action Name:	Operator aligns one train of Low Pressure Injection System for closed-loop RHR cooling
Human Action Identifier:	H3RHR1
Event Tree:	SBLOCA
General Description:	The task is to switch off High pressure Injection Pump and to provide Residual Heat Removal Cooling
Scenario:	1. Reactor scram actuates by: L_P < 4000 mm; P_I < 150 kgf/cm^2; D t_S < 10°C/h 2. One train of High Pressure Injection System is available 3. Steam Dump to the Atmosphere and Emergency Feedwater are available 4. Operator decreases pressure down to 18 kgf/cm^2
Task:	On pressure level 18 kgf/cm^2, shown by annunciation in MCR (or on monitors or on pressure meters), operator should switch off HPIS and to align LPIS through line for RHR cooling. It is complex action, it includes sequential switching of several switches for valves of TQ40 and TQn4, which are o MCR panels. These actions follow instruction.
Time Window:	>30 min after initiating event
Difficulty (stress)	H
Consequence of actions	second after first success
HEP:	0.0161

Scenario Sheet for Human Action H3RHR2

Human Action Name:	Operator aligns one train of Low Pressure Injection System for closed-loop RHR cooling
Human Action Identifier:	H3RHR2
Event Tree:	SBLOCA
General Description:	The task is to switch off High pressure Injection Pump and to provide Residual Heat Removal Cooling
Scenario:	1. Reactor scram actuates by: L_F < 4000 mm; P_I < 150 kgf/cm^2; D t_S < 10°C/h 2. One train of High Pressure Injection System is available 3. Steam Dump to the Atmosphere and Emergency Feedwater are not available 4. Hydroaccumulators inject into the RCS 5. Operator decreases pressure down to 18 kgf/cm^2
Task:	On pressure level 18 kgf/cm^2, shown by annunciation in MCR (or on monitors or on pressure meters), operator should switch off HPIS and to align LPIS through line for RHR cooling. It is complex action, it includes sequential switching of several switches for valves of TQ40 and TQn4, which are o MCR panels. These actions follow instruction.
Time Window:	>30 min after initiating event
Difficulty (stress)	H
Consequence of actions	second after first success
HEP:	0.0161

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BG9600414

INTEGRATED COMPUTER CODES FOR NUCLEAR POWER PLANT SEVERE ACCIDENT ANALYSIS

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INRNE - BAS

1. INTRODUCTION

Since the accident at Three Mile Island Unit 2 (TMI-2) increased attention has been given to the details of accidents that could take place at a nuclear power plant and that could result in radioactive releases to the environment in excess of accepTable levels. Because of the scale and complexity of the relevant postulated events, it is not possible to carry out direct experimental investigations to characterise them. Instead, these complex sequences have been modelled as a set of simpler component phenomena. Experimental programmes have been undertaken to characterise the results of occurrences of possible combinations of these phenomena in postulated events. The improved understanding of the various individual phenomena has been incorporated into analytical computer codes, thus synthesising the individual phenomena into sequences of events representing hypothetical accidents.

2. MELTDOWN ACCIDENT ANALYSIS PROGRAM

The Modular Accident Analysis Program (MAAP) is a computer code which simulates light water reactor system response to accident initiation events [1]. It is prepared as a part of the IDCOR (Industry Degraded COre Rulemaking) program to investigate the physical phenomena which might occur in the event of a serious light water reactor accident leading to core damage, possible reactor pressure vessel failure and possible containment failure and depressurization.
MAAP includes models for the important phenomena which might occur in a serious light water reactor accident. There are two parallel versions of MAAP, MAAP/BWR and MAAP/PWR, for the two general light water reactor types in use. MAAP can predict the progression of hypothetical accident sequences from a set of initiating event to either a safe, sTable, coolable state or to containment failure and depressurization. MAAP treats a wide spectrum of phenomena including steam formation, core heatup, cladding oxidation and hydrogen evolution, vessel failure, corium-concrete interactions, ignition of combustible gases, fluid (water and corium) entrainment by high velocity gases, and fission product release, transport and deposition. MAAP treats all of the important engineered safety systems such as emergency core cooling, containment sprays, fan coolers, and power operated relief valves, and the auxiliary or reactor building can be modelled for sequences in which it is important. In addition, MAAP allows operator interventions and incorporates these in a very flexible manner, permitting the user to model operator behaviour in a general way. The user models the operator by specifying a set of variable values and/or events which are the operator interventions conditions. There is a large set of actions the operator can take in response to the intervention conditions.

The user may establish one or more intervention conditions by specifying limits for any of a set of key variables or by declaring any of more than 100 event flags as key events. When a key variable reaches its specified limit, or a key event flag changes status, program execution pauses and operator actions, also specified by the user, are taken. The operator actions consist of changes to event flags numbered 200 and above. In this way, MAAP is directed by a pseudooperator who uses present plant conditions to make operational decisions.

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Periodically during a transient, MAAP writes restart files. This allows the user to make a subsequent run starting at any time covered by the original run. This provides additional flexibility for the user to simulate operator actions or to change external events. Restart data files are written at time intervals chosen by the user and whenever a program interventions occurs. MAAP can then resume execution from any of the times at which a restart file entry was written. The restart can have a new program intervention conditions, new operator actions, and even changes to the input parameter file.

MAAP has a modular structure in which separate subprograms are dedicated to specific region models and physical phenomena. This facilitates changing the code because improvements to phenomenological or region models can be made relatively small subprograms. Each MAAP version, MAAP/BWR and MAAP/PWR, consists of a main program which directs program execution through several high level subroutines. Depending on the containment/primary system design, the program calls a sequence of system and region subroutines at each time step. These subroutines call, in turn, phenomenology subprograms as required. At the lowest level, a set of property-library subprograms are available to provide physical properties.

The MAAP code uses a two stage computational procedure in which the present values of the dynamic variables describing the state of the system (often masses and internal energies) are used to calculate their rates of change. Then the rates are integrated in a separate subroutine to provide updated values of the dynamic variables. The integration technique used during the development of the MAAP is an explicit, first order, Euler integration. An alternative method, a second order Runge-Kuta integration can be selected through the parameter file. The differences are not great; a Loss-of-All-Power PWR example transient using the second order method had a running time twice as long as that of the first order method. Key procedure events (e.g., time to reactor vessel failure) were within a few percent of each other generally much closer. A typical MAAP run has time steps as small as 0.005 seconds and as large as 20 seconds. MAAP has been written to execute both in a batch mode and in an interactive mode where the user monitors the execution at a terminal. The batch and interactive modes are entirely parallel and use the same input desks (file of card images). They differ only in method and timing of entering the cards (lines) of the input desk (file).

MAAP can model an auxiliary building which receives the discharges from either the containment or the primary system. The auxiliary building model can be run simultaneous with the primary system and containment models or in a stand-alone model. In the later, the code reads an input generated in an earlier MAAP run that supplies boundary conditions to the model.

The parameter file, required by MAAP to define the reactor system, consists mainly of plant-specific data which will not change from one run to another. These are relegated to a disk file which is read by MAAP at the start of execution. Accident specific inputs, such as accident initiators and operator actions, are contained in a separate input desk which is read by MAAP during execution. The user may change parameter file entries for individual runs by specifying those changes in the input desk. Thus, the parameter file for a specific plant needs to be prepared only once and temporary changes to any parameter entries can be made at execution time without manipulating the parameter file itself. An accident summary is printed at the end of a run and provides a chronology of significant events such as engineered system responses and operator actions.

MAAP has been issued in several versions since about 1982. The first version receiving wide distribution was MAAP 2.0B.

3. ICARE COMPUTER PROGRAM

The ICARE computer code is being developed at the Institute for Protection and Nuclear Safety of CEA (France). This analytical work [2] is a part of the Severe Fuel Damage program [3] conducted in France in order to clearly understand the main physical mechanism occurring during the core degradation of PWR.

ICARE models the progression of reactor core damage including: core heat-up, loss of geometry by melting and embrittlement, relocation of materials, crucible formation and fission products release. It works as a stand-alone code to describe both experimental facilities and reactor core in severe situations; in case of reactor calculations, ICARE uses boundary conditions provided by the French Reactor Coolant System CATHARE code [4].

During a severe accident on PWR, hydrogen and heat are produced by oxidation of zircaloy in fuel claddings and absorber rod guide tubes as well as oxidation of stainless steel in absorber rod claddings and vessel structures. The prediction of the right amount of hydrogen and heat is of great importance for the analysis of containment integrity in case of loss of coolant from primary circuit and for the analysis of whole degradation respectively.

ICARE2 code (version 2) uses a new data organisation technique allowing especially a dynamic management of the computer memory, called SIGAL (in French: Structure Informatique d'Accueil et de Gestion de Logiciels).

This technique sets up several functions concerning in the same time developers and users. Each function is associated with either a library or a code:

- a dynamic memory management library to optimise the use of central computer's memory;
- a memory organisation library to handle and modify SIGAL data bases stored in the central memory;
- a data reader;
- a data checker;
- an analyser program to interpret special user's instructions in order to perform all manipulations of SIGAL data bases;
- a self governing graphic program called T.I.C. (in French: Trace Interactif de Courbes). A general overview of the ICARE2 architecture is given in figure 1.

The modular structure of the code allows many types of topologies to be described: PHEBUS SFD and FP, PFB, CORA, PWR core.

The main program calls in series different modules which are associated to specific physical tasks: fluid dynamics, thermics, flow down and relocation, chemical reactions, ect. The numerical scheme of the main physical modules is as implicit in time as possible, but the coupling between them is generally explicit. Therefore, the energetic error induced by coupling between hydraulic and thermic modules controls essentially the global time step management.

The data exchanges between each of these modules are performed through SIGAL data base. This logic, combined with the modular structure of the code, makes it possible new modules to be developed independently and after self-governing tests to be easily introduced in ICARE2.

ICARE2 is composed of about 30 000 statements entirely written in FORTRAN 77 and, therefore, can be implemented under any system compatible with the FORTRAN 77 language. However, the SIGAL software must be adapted according to specific logic of the different hardware (IBM, CRAY, VAX, SUN, ect.).

ICARE2 is implemented in France on IBM, CRAY and SUN computers. It has been released to SPAIN and to ISPRA.

The ICARE2 input file is presented in the form of independent blockdata, each of them defining various types of quantities such as metallic structure geometry, boundary conditions, physical exchanges, computation options (time step, graphic or restart storages, ect.). There is no compulsory order for blockdata description in the input file and no predefined relationships between each blockdata. The latter point, which gives large flexibility to introduce data, allows users to perform very easily sensitivity studies. The input data acquisition is performed by a data reader which is able to read structured user's data, to check their syntax and to store automatically them in a SIGAL data base. The data checker compares user's data with a predefined data base, called control file. This data base consists of a file created by developers and containing the correct data form as well as the physical or logical elementary rules they have to respect (such as data compulsory aspects, blockdata variable types, physical pertinence of boundary conditions, number of terms in Tables, increasing or decreasing order of these terms, ect.).

4. SOURCE TERM CODE PACKAGE

The Source Term Code Package (STCP) is a set of computer codes which allows analyses of nuclear reactor accidents to produce predictions of fission product release to the environment as a function of reactor design and specifications of the assumed accidents [5]. In figure 2 is shown the flow diagram of the STPC. The overall thermal-hydraulics is provided by the MARCH-3. Release of fission products and aerosols during core-concrete interactions is predicted with the VANESA code. Detailed thermal-hydraulics and fission product transport in the reactor coolant system are provided by the TRAP-MELT3. Finally, fission product transport in the containment is predicted by NAUA code.

MARCH-3 describes the behaviour of the reactor during a severe accident and, with approximately 185 routines, is the largest of the codes involved in the STCP. The MARCH-3 code evaluates the following phenomena:

1. Heatup of the reactor coolant inventory and pressure rise or safety valve settings with subsequent boiloff;
2. Initial blowdown of the of the coolant from the reactor coolant system;
3. Generation and transport of heat within the core, including boiloff of water from the reactor vessel;
4. Heatup of the fuel following core uncovery, including the effects of metal-water reactions;
5. Melting and slumping of the fuel onto the lower core support structures and into the vessel bottom head;
6. Interaction of the core debris with residual water in the reactor vessel;
7. Interaction of the core debris with the reactor vessel bottom head;
8. Interaction of the core debris with the water in the reactor cavity;
9. Attack of the concrete baseman by the core and structural debris;
10. Relocation of the decay heat source as fission products are released from the fuel and transported to the containment;
11. Mass and energy additions to the containment associated with all foregoing phenomena and their effects on the containment temperature, pressure, and steam condensation;

12. Effects of burning of hydrogen and carbon monoxide on the containment pressure and temperature;
13. Leakage of gases to the environment.

For MARCH-3, input is required to describe plant, to select modelling options, to describe control parameters for safety system operation.

The MARCH-3 output provides a wide variety of information on the thermal and hydraulic conditions in the reactor system, as well as the containment, throughout the course of the accident sequence being analysed. Among the key outputs are the timing of containment events, status of the core, and pressure and temperature in the containment. Additionally, extensive detail is available on core and structure compositions, distribution of water with in the entire system, mass and energy balance audits, ect.

The VANESA model is a mechanistic description of the aerosol generation and fission product release during core debris interactions with concrete. The model predicts the mass, composition, and meanparticle size of radioactive and non-radioactive materials liberated as vapors or particles during interactions. The model indicates whether mass release is by vaporisation processes or mechanical processes. VANESA takes the CORCON output of the MARCH-3 and models the reduction of the H_2O and CO_2 to H_2 and CO, as well as the loss of other materials from the pool as aerosols. The gas release from the core-concrete interaction is an important part of most accident sequences because it provides a severe load on the containment at the same time that a large amount of airborne material is being produced. The rate of deposition of vapors and aerosol moving through structures of reactor coolant system (e.g., the upper plenum or the pressurizer) can be calculated given the temperature of these structures and the flow rate, composition, and temperature of the gas. The MERGE code provides the required flow rates, gas conditions, and temperatures. TRAP-MELT2 code, which runs as a subroutine in TRAP-MELT3, handles ten species of materials, including noble gases in the STCP. The noble gases are not retained in the reactor coolant system and are thus considered in TRAP-MELT only for decay heat calculations and for bookkeeping purposes. The three species, CsI, CsOH and Te account for all the volatile fission products of interest in a TRAP-MELT calculations. These three forms are treated as vapors as they leave the core: they can condense an walls and aerosol particles, evaporate from were they have condensed, or become attached to wall surface by some chemical or physical mechanism (sorbtion).

Some portion of the aerosol produced in the vessel during core heatup and those ex-vessel aerosols produced during the core concrete interaction eventually arrive in the reactor containment structure. Natural processes of agglomeration and deposition lead to retention of aerosols within the containment. In addition, some containment structures are equipped with water sprays, ice condensers, or water suppression pools, which cause further retention of aerosol and fission products. The NAUA-MOD4 computer code is being used for analysing these effects.

5. EXPERIENCE OF STCP IMPLEMENTATION

5.1. VAX SYSTEM IMPLEMENTATION

STCP Mod.1 was implemented on a VAX/VMS computer system. The package was run on a 32-bit supper-minicomputer VAX 11/150 with VAX/VMS V4.7 operating system. Using the 192 version of the programme MARCH3, the TMLB sample problem for Zion Unit 1 NPP (PWR with a large, dry containment) was calculated successfully. However, when analysing the TMLB scenario for a VVER-440 (V 213) NPP, MARCH3 failed to calculate the entire sequence due to loss of accuracy. In order to remove this obstacle, a DOUBLE

PRECISION VAX-version MARCH3D was created. In this version, the CPUSSEC function was recoded to return the elpased real time through substitution the CDC intrinsic function SECOND by the VAX/VMS intrinsic function SECNDS. The last one returns the time of the day. Using the created VAX-version MARCH3D, the TMLB scenarios for Zion and VVER-440 NPP were also calculated.

Table 1 shows the chronology of the main accidents events during the Zion TMLB accidents calculated by MARCH3 and MARCH3D an a VAX 11/750 computer. For a comparison, the sample output results for the same scenario, provided by the Battelle Columbus Laboratories, are also given in Table 1.

In order to estimate the influence of the VAX 11/750 word length representation on the results obtained, the relative deviations of some Zion TMLB MARCH3- and MARCH3D-calculated parameters from the sample values were calculated. The relative deviation of the parameters, dev_i, was calculated according to:

$$dev_i = \frac{P_{calc,i} - P_{sample,i}}{P_{sample,i}} 100, \%, \qquad (1)$$

where

$P_{calc,i}$ is MARCH3- or MARCH3D-calculated value of the parameter at the moment of accident event i;
$P_{sample,i}$ - sample value of the parameter at the moment of accident event i.

Figures 3 and 4 show the relative deviations of the Zion TMLB MARCH3- and MARC3D-calculated reactor containment pressure and debris temperature as sample accident time functions.

TABLE 1 - CHRONOLOGY OF THE MARCH3- AND MARCH3D-CALCULATED ZION TMLB ACCIDENT EVENTS ON A VAX 11/750 COMPUTER

No	ACCIDENT EVENT	TIME (min)		
		SAMPLE	MARCH3D	MARCH3
1	Accident initiation	0.00	0.00	0.00
2	Steam generator dryout	93.30	93.30	95.65
3	Core uncovery	126.63	126.63	126.23
4	Start of core melting	149.13	149.13	148.98
5	Core slumping	168.13	168.13	167.48
6	Bottom head heatup	169.88	169.88	169.23
7	Bottom head failure	178.57	178.57	177.92
8	Start of water-debris interaction	178.57	178.57	177.92
9	Containment failure	279.61	179.61	178.96
10	Reactor cavity dryout	242.18	242.04	239.42
11	Start of debris-concrete interaction	307.18	304.04	305.42
12	Normal exit	907.18	907.04	905.42

5.2 IBM SYSTEM IMPLEMENTATION

STCP Mod.1 was also implemented on an IBM computer system. The package was run on a 32-bit IBM 3081 computer. The software included MVS/XA operating system, ISPF utility an VS-FORTRAN V-1.3 compiler. During the VS-FORTRAN compilation of the origin MARCH3 (V 192) code, some warnings (but not syntax errors) were encountered. Zion TMLB sample problem wad run on the machine and it was noted that the resultant output contained some differences from the sample output results. Consequently, the second attempt was to recoded MARCH3 into DOUBLE PRECISION. Repetitive checking of source lines was necessary during the conversion. As a result of these modifications, the full functionality of the code was retained. Using the created IBM DOUBLE PRECISION version MARCH3D, the TMLB scenarios for Zion and VVER-440 (V 213) NPPs were calculated.

Table 2 shows the chronology of the main accident events during the Zion TMLB accident sequence calculated by the MARCH3D DOUBLE PRECISION versions on a VAX 11/750 and on a IBM 3081 computers. For a comparison, the sample problem output results for the same scenario, are also given in Table 2. The figures 7 to 10 show a very good coincidence between the results from calculation and the sample problem output results.

In order to estimate the influence of the computer used on the results obtained, the relative deviations of some Zion TMLB VAX 11-750- and IBM 3081-calculated parameters, using the created MARCH3D DOUBLE PRECISION versions, from the sample output values were calculated according to Equation (1).

Figures 5 and 6 show the relative deviations of the Zion TMLB VAX 11/750- and IBM 3081-calculated reactor containment pressure and debris temperature as a function of the sample accident time.

TABLE 2 - CHRONOLOGY OF THE MARCH3- AND MARCH3D-CALCULATED ZION TMLB ACCIDENT EVENTS ON A VAX 11/750 COMPUTER

No	ACCIDENT EVENT	TIME (min)		
		SAMPLE	VAX	IBM
1	Accident initiation	0.00	0.00	0.00
2	Steam generator dryout	93.30	93.30	93.30
3	Core uncovery	126.63	126.63	126.63
4	Start of core melting	149.13	149.13	148.13
5	Core slumping	168.13	168.13	168.13
6	Bottom head heatup	169.88	169.88	169.88
7	Bottom head failure	178.57	178.57	178.57
8	Start of water-debris interaction	179.57	178.57	179.57
9	Containment failure	179.61	179.61	179.61
10	Reactor cavity dryout	242.18	242.04	240.30
11	Start of debris-concrete interaction	307.18	307.04	306.30
12	Normal exit	907.18	907.04	906.30

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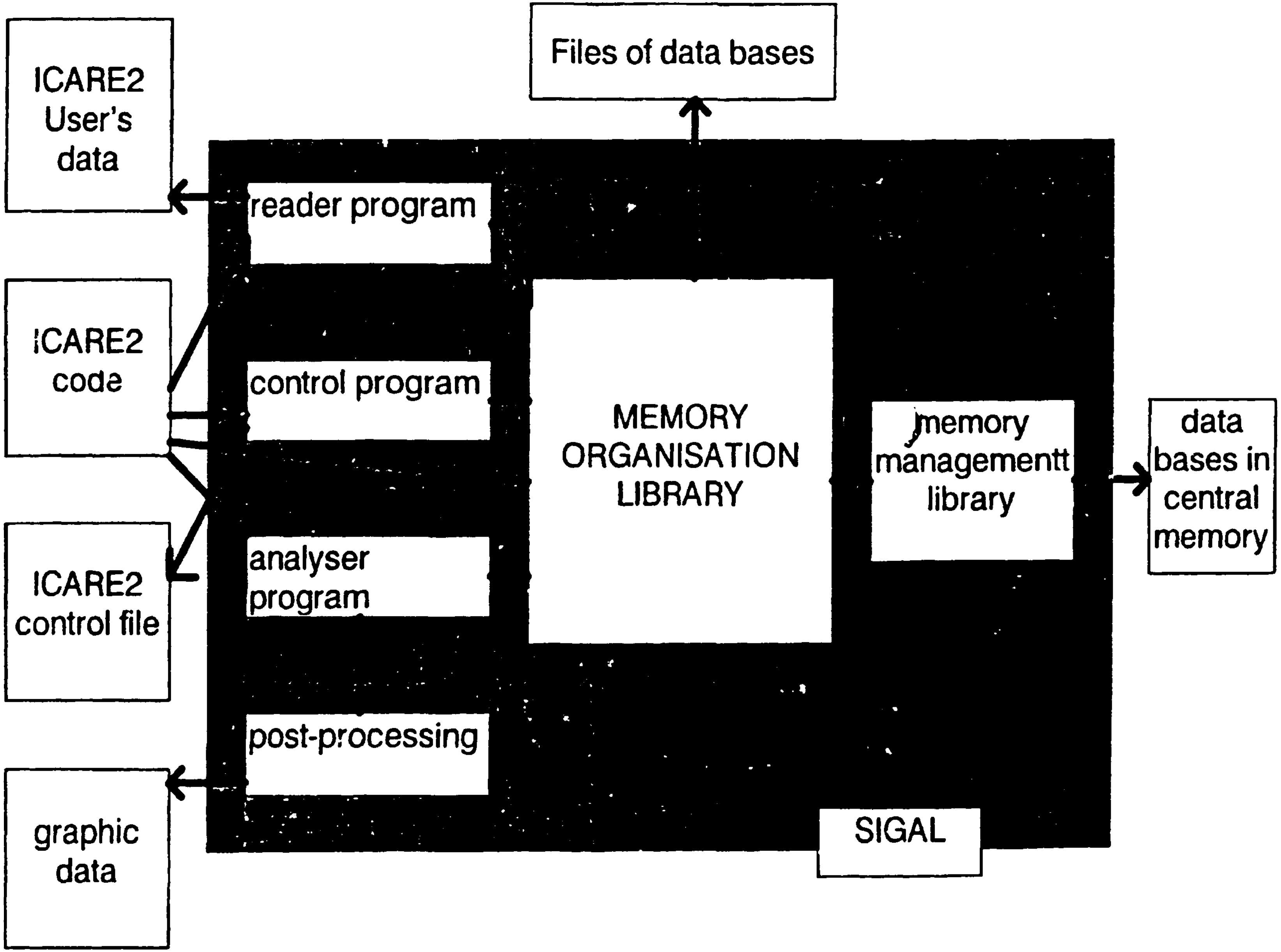
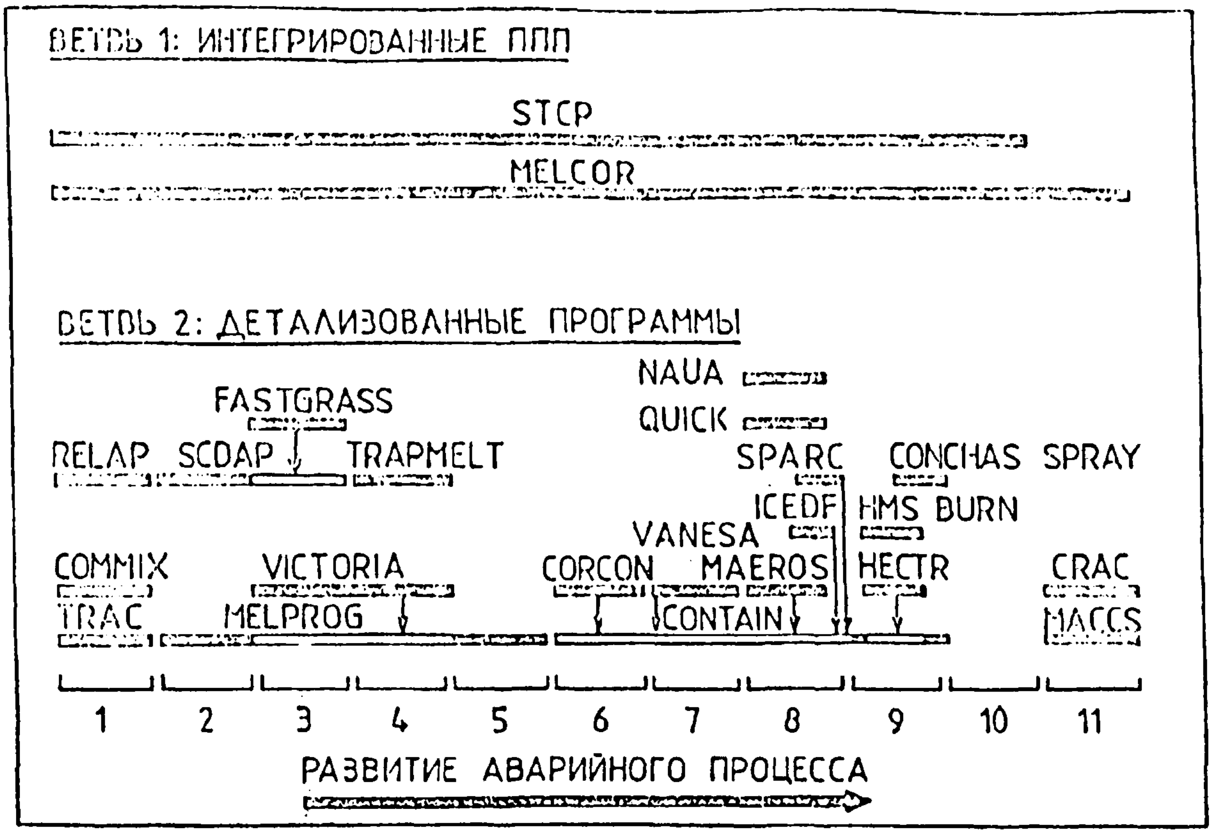


Fig. 1 - A general overview of the ICARE2 architecture



1 - Thermal Hydraulics

2 - Core Melting

3 - Release Frome Fuel

4 - Transport in Reactor Coolant System

5 - Vessel Failure

6 - Concrete Interactions

7 - Release from Debris

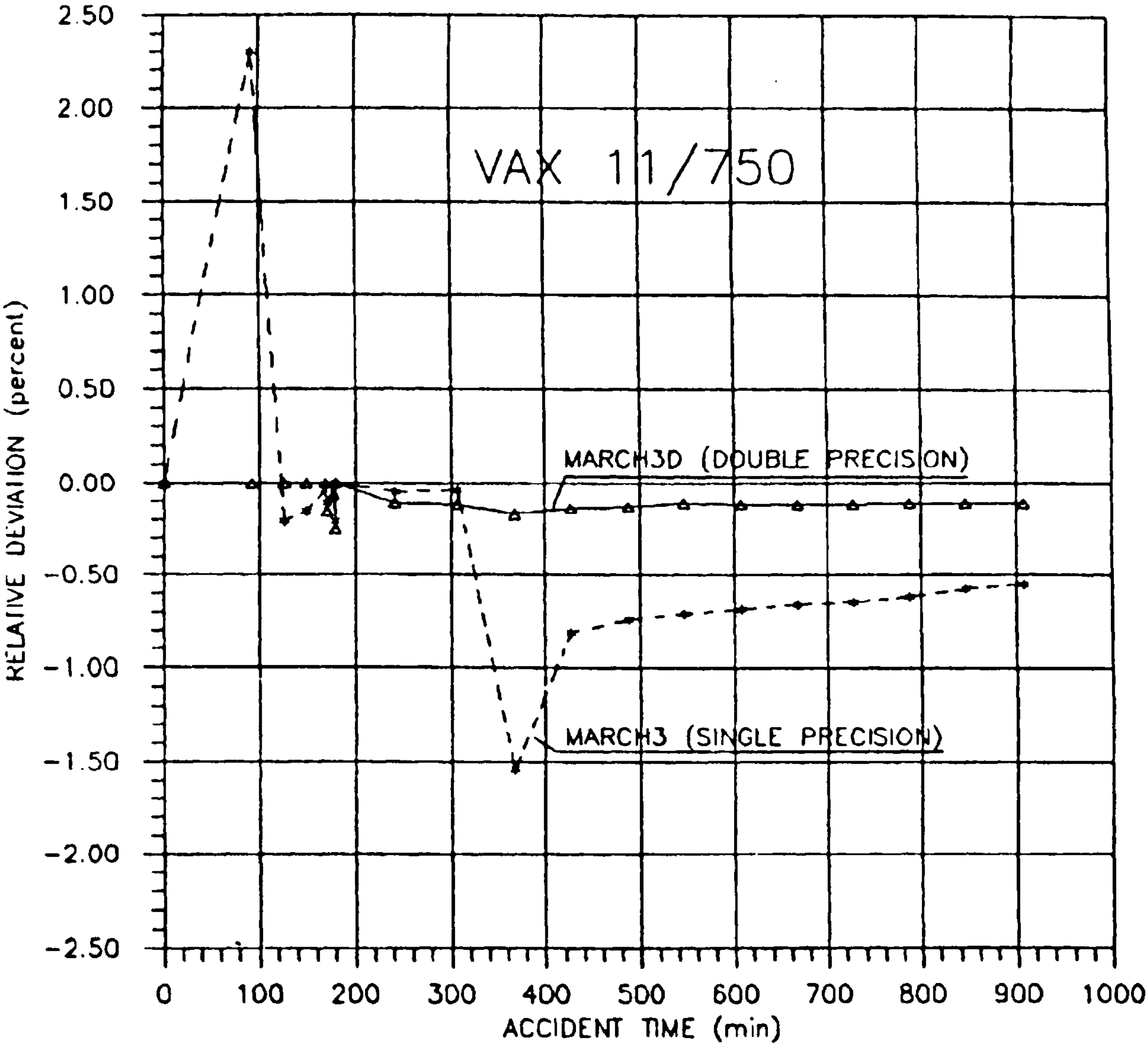
8 - Transport in Containment

9 - Containment Loads

10- Containment Performance

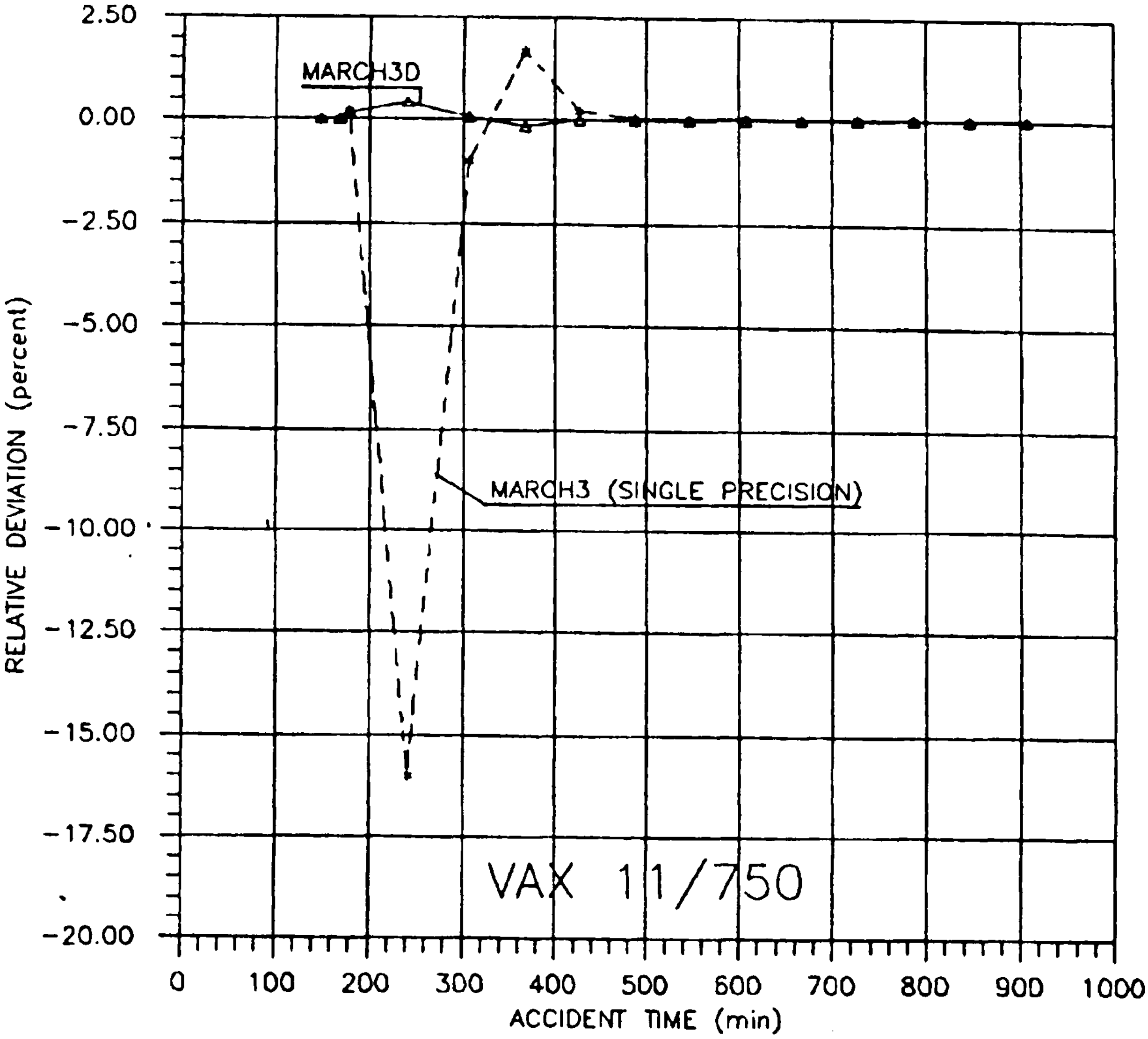
11- Off Site Consequences

Fig. 2 - Flow diagram of the STCP



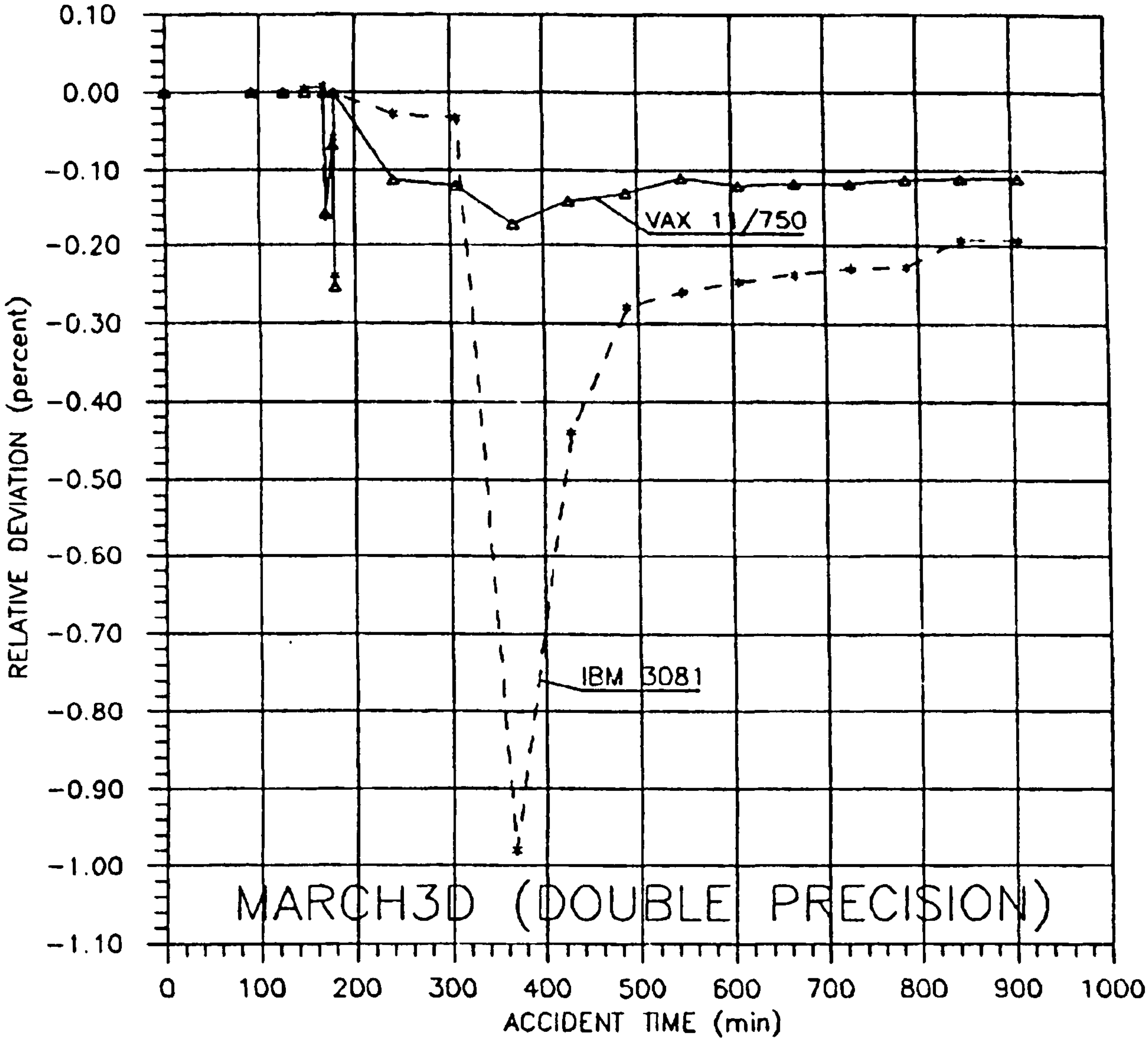
RELATIVE DEVIATION OF THE MARCH3- AND MARCH3D-
CALCULATED ZION TMLB CONTAINMENT PRESSURE
ON A VAX 11/750 FROM THE SAMPLE VALUES

Fig. 3 - Containment pressure



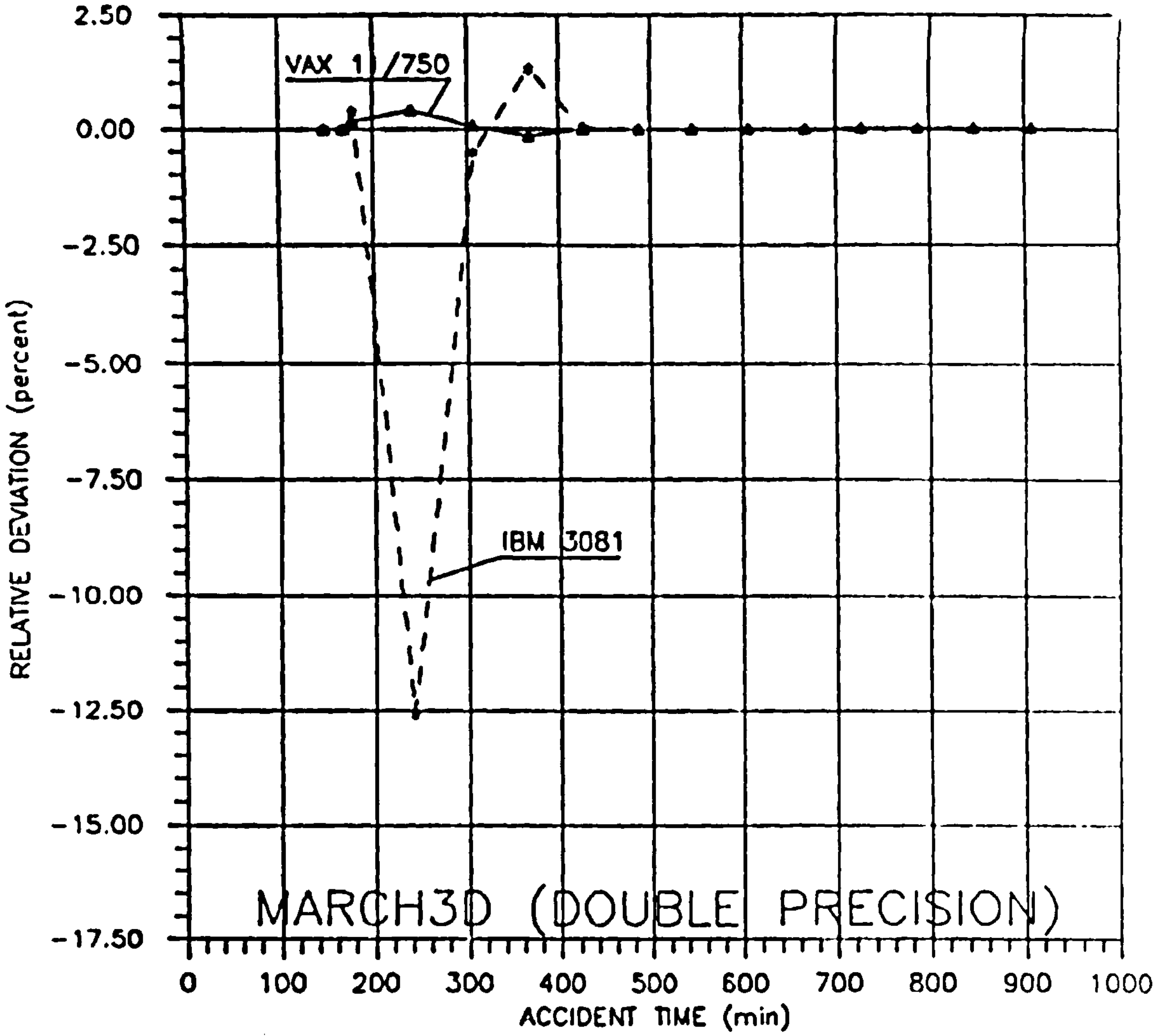
RELATIVE DEVIATION OF THE MARCH3- AND MARCH3D-
CALCULATED ZION TMLB DEBRIS TEMPERATURE
ON A VAX 11/750 FROM THE SAMPLE VALUES

Fig. 4 - Debris temperature



RELATIVE DEVIATION OF THE MARCH3D-CALCULATED ZION TMLB CONTAINMENT PRESSURE ON VAX 11/750 AND IBM 3081 COMPUTERS FROM THE SAMPLE VALUES

Fig. 5 - Containment pressure



4 RELATIVE DEVIATION OF THE MARCH3D-CALCULATED ZION TMLB DEBRIS TEMPERATURE ON VAX 11/750 AND IBM 3081 COMPUTERS FROM THE SAMPLE VALUES

Fig. 6 - Debris temperature

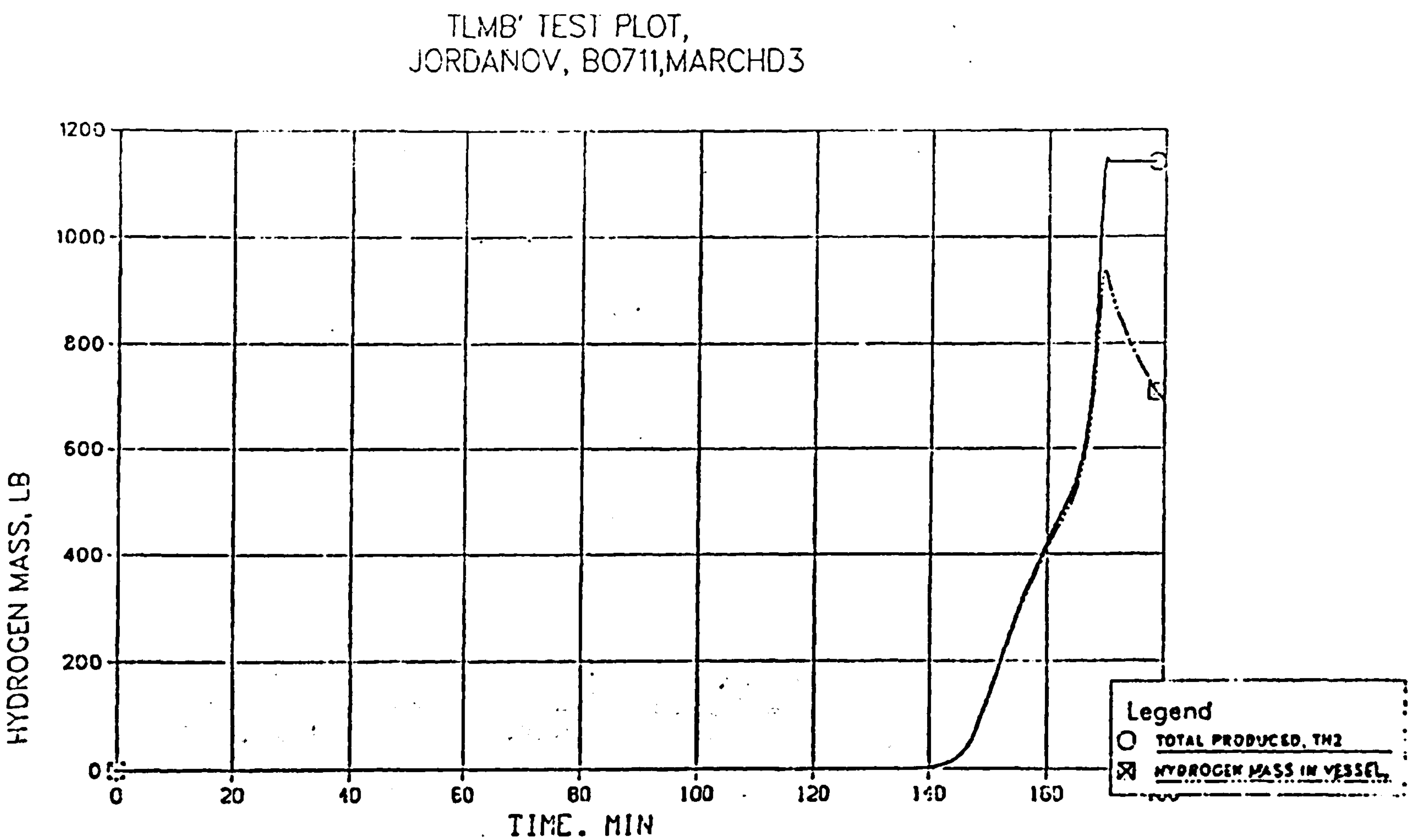


Fig. 7 - Hydrogen Mass, LB

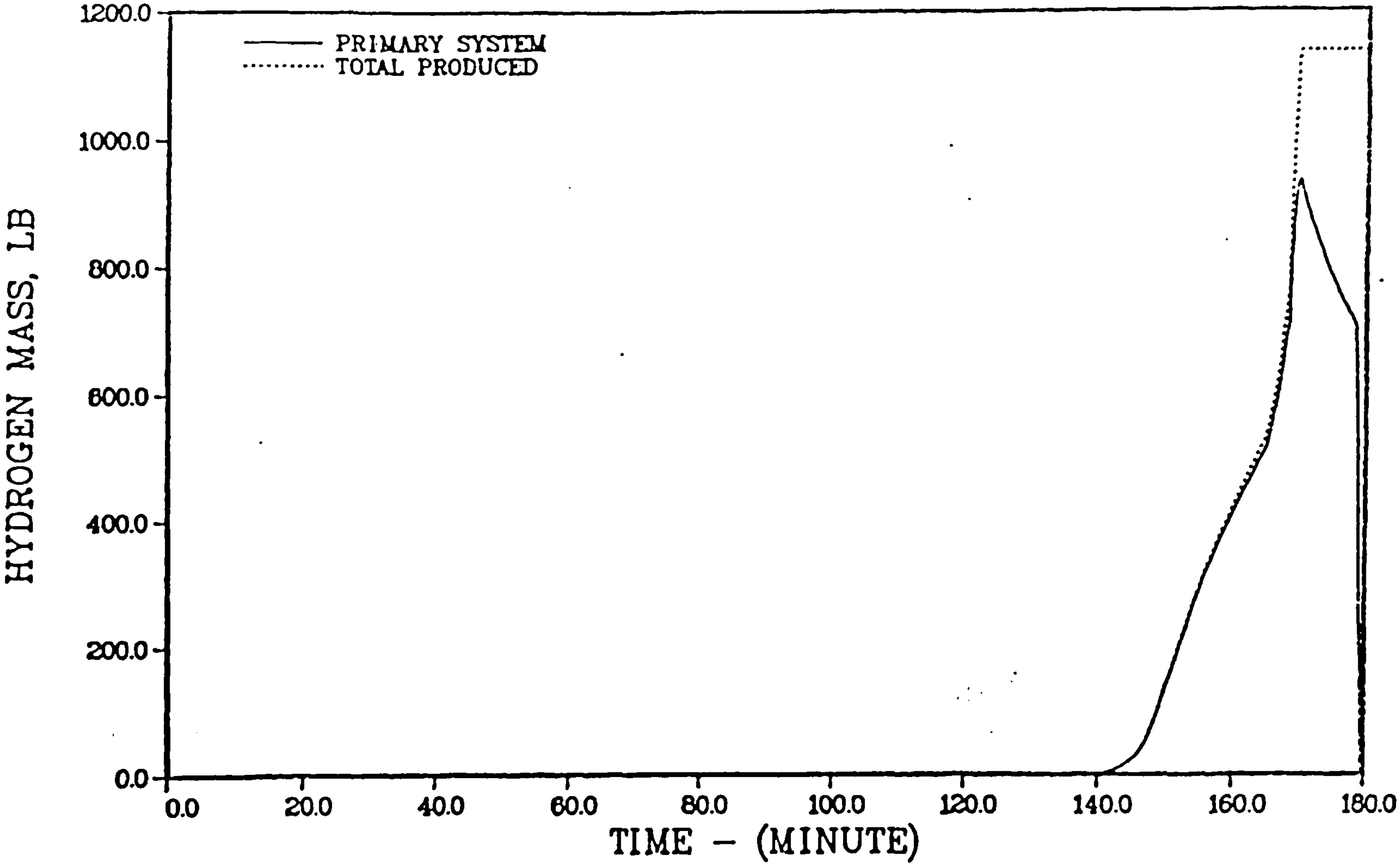


Fig. 8 - Hydrogen Mass, LB

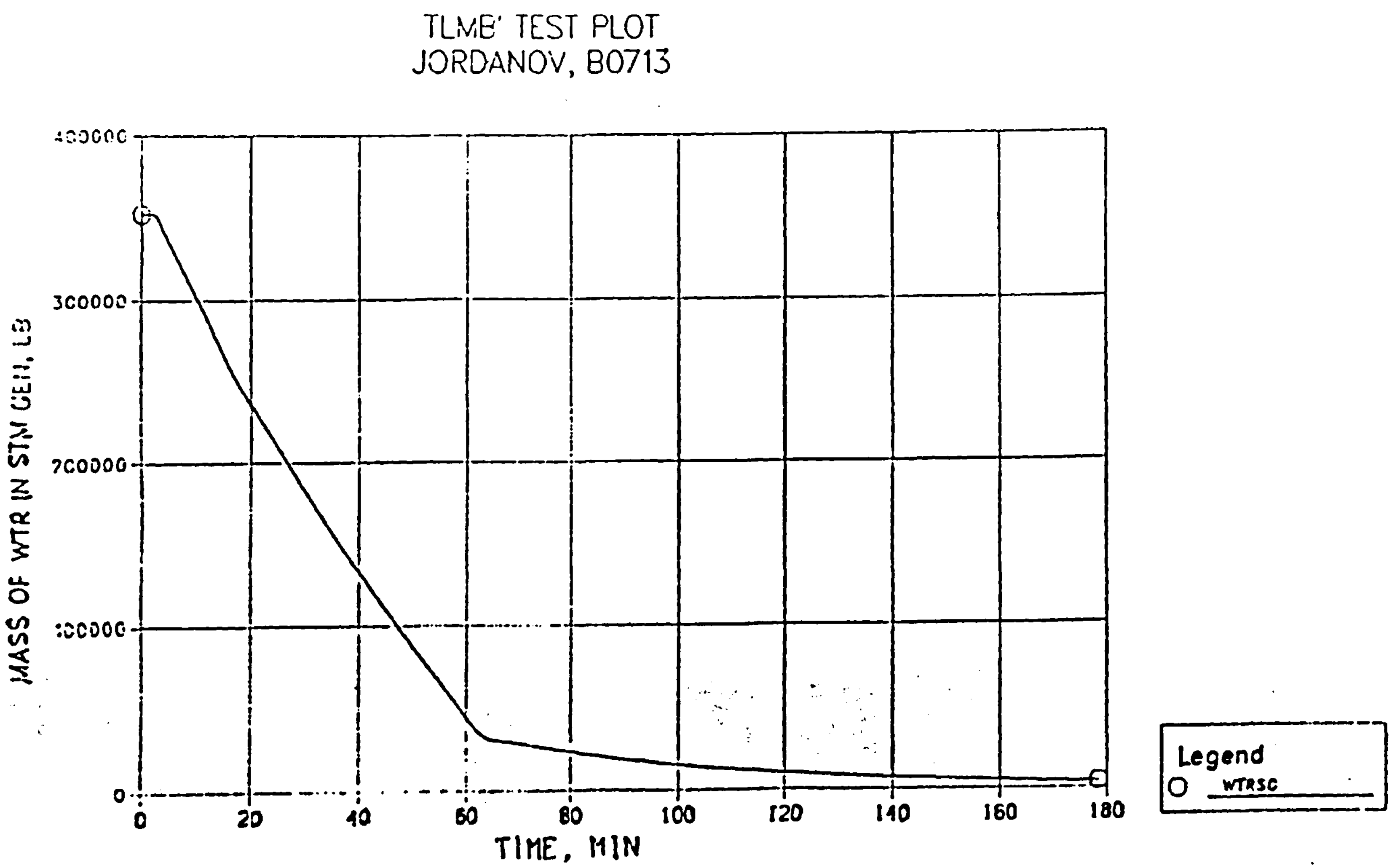


Fig. 9 - Mass of WTR in STM GEN, LB

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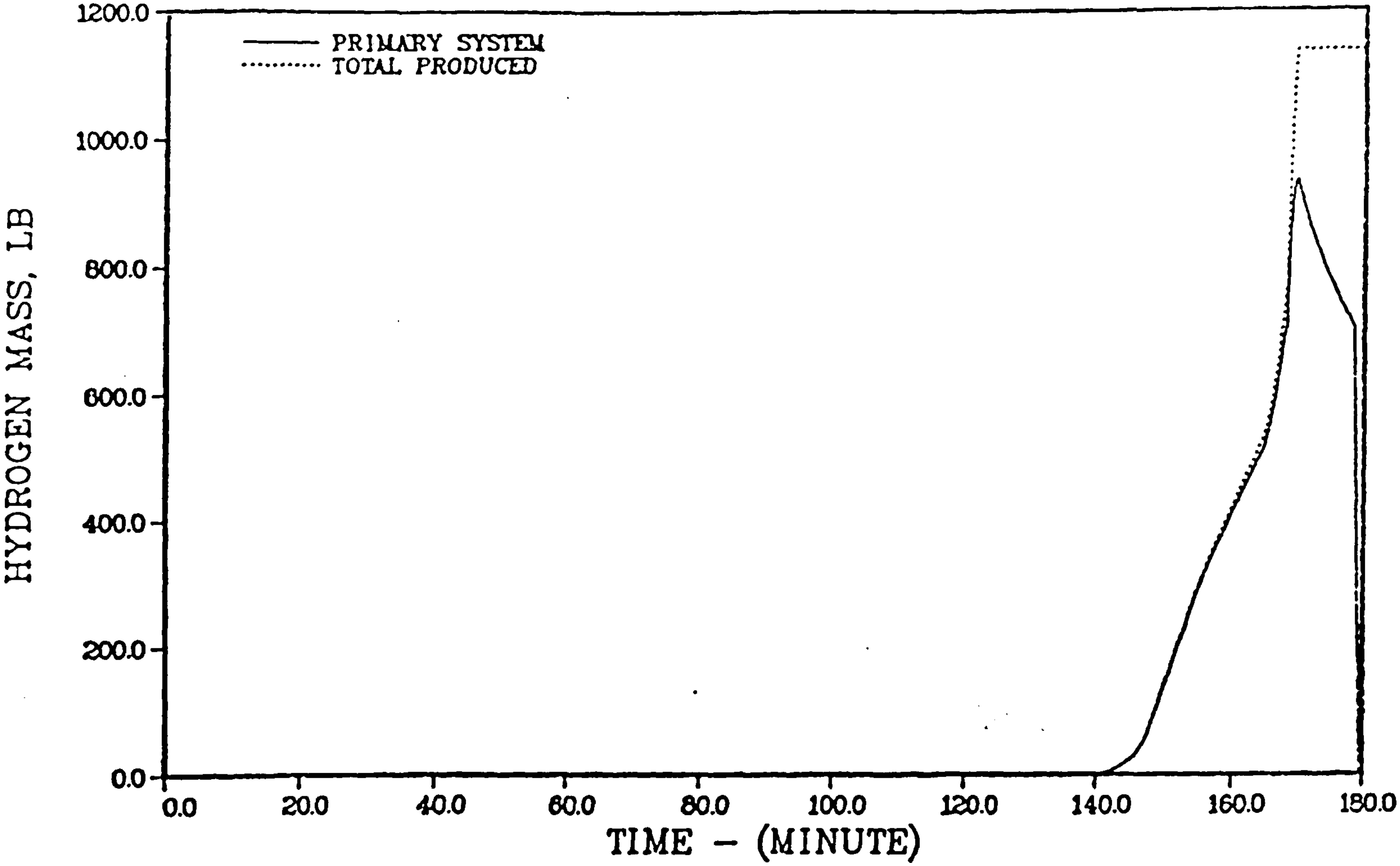

Fig. 10 - Mass of WTR in STM GEN, LB

BG9600415

"KOZLODUY" NPP WWER - 440/230 REACTOR PRESSURE VESSEL RADIATION LIFE TIME

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The designed life time of Kozloduy nuclear units WWER440/230 is 30 years. The world practice shows that as a result of fast neutron bombardment in the reactor pressure vessel (RPV) metal a process of embrittlement is running. As result the radiation life time gets less than the designed life time. For the moment there are relatively accurate empirical methods for predicting the range of neutron irradiation embrittlement (NIE) of the RPV metal while the embrittlement rate law of neutron re-irradiation embrittlement (NRE) after annealing is not definitely established. In spite of different procedures and reconstruction for neutron embrittlement effect mitigation the extension of life time for RPV with high P and Cu content in the weld metal is under safety limit. The possibility for solving of this problem is the determination of re-embrittlement rate law. In this moment the conservative re-embrittlement law is accepted in the world standards for assessment of RPV integrity. Now there are new data supporting the model for "lateral (horizontal) shift" of the critical transition temperature curve after neutron re-irradiation [1,2,3]. This model gives priority over conservative law as the re-embrittlement rate decrease gives extension of radiation life time. Those data are not statistically well grounded and a future confirmation of their validity is necessary.

The aim of this work is to compare the RPV WWER440 NPP "Kozloduy" radiation life time, calculated by means of the different re-embrittlement rate laws after annealing using updated parameters describing neutron irradiation embrittlement and standard method for RPV integrity assessment.

1. Activities for mitigation of the neutron embrittlement of RPV metal

In order to increase the life time of RPV some activities for mitigating the rate of NIE, restoring the mechanical properties and restricting the possibility for thermochocks are performed in NPP Kosloduy. The most important of them are:

- Decreasing the neutron loading on RPV metal by means of installation of 36 dummy elements in the periphery of the core zone. The years of dummy element loading are given in table 1:

Table 1

	Unit 1	Unit 2	Unit 3	Unit 4
Year	1987	1988	1987	low leakage

- Heating the water in tank for emergency core cooling up to 55°C.
- The recovery annealing is conducted according Russian methods : 475°C/150 h, heating rate <20°C and cooling rate <30°C as follows:

Table 2

	Unit 1	Unit 2	Unit 3
Year	1989	1992	1989

- Actualisation of the operation instruction for pressure decreasing in the case of compensated primary leak running at high pressure in RPV.
- ISI before and after annealing

- Actualisation of the P-T start and shut down diagrammes after annealing and of the permissible temperatures for hydrotests.
- Starting in 1992 a programme for installation of fast acting valves in main steam piping.

2. RPV metal embrittlement criteria parameters

For calculation of critical temperature of embrittlement (Tkf) a number of parameters, describing the metal properties, the irradiation conditions and the neutron field are necessary. Unfortunately in the early years of nuclear power production the process of NIE has not been investigated well enough and the values of some parameters haven't been measured and registrated during manufacture.

According Russian standards [4] the critical temperature of ductile to brittle transition (Tkf) for weld metal is given by:

$$Tkf = Tko + Af\ (F/Fo)^{0.33} = Tko + 800(P + 0.07Cu)\ (F/Fo)^{0.33} \quad (1)$$

where:

Tko- critical temperature of ductile to brittle transition for non irradiated metal, Af - chemical coefitient of embrittlement, F - fluence, $Fo = 10^{18}$ n/cm^2.

2.1 Chemical composition

With respect to neutron induced embrittlement of RPV WWER440/230 metal only the P and Cu content in weld 4 is decisive. The P and Cu concentrations are shown in table 3. The Unit 3 and 4 values are factory data. The data for Unit 2 are received lately by means of optical emission spectroscopy and wet chemical analysis of templet material taken out from RPV [5] and for Unit 1 - by scraps chemical analysis . While the P an Cu contents in weld 4 -Unit 2 coincide with the predicted ones, those for Unit 1 are quite different: the Cu content is lower and P content - extremely higher.

Table 3

	Unit 1	Unit 2	Unit 3	Unit 4
P , %wt.	0.05	0.037	0.036	0.021
Cu, %wt.	0.11	0.18	0.20	0.04
A_f	48.3	40.1	40.0	19.0

2.2 Initial critical temperature of embrittlement (Tko)

Similarly to the impurity content, the Tko values have been determinated in factory only for Unit 3 and 4 (table 4). For Unit 2 Tko has been determinated by means of high temperature annealing (T=560°C/ 2h) of subsize specimens manufactured from templets material and for Unit 1 recalculated with the new P content . The increasing of P content up to 0.051% increase the Tko value from 52°C to 65°C [6]. There is an uncertainty in the last value due to the inaccuracy of formulae used for calculation.

Table 4

	Unit 1	Unit 2	Unit 3	Unit 4
Tko, °C	65	50	50	5

2.3 Residual part of Tkf shift (Δ Tres) and re-embrittlement law

After annealing the Tkf increase can by determinated by standard conservative method :

$$Tkf = Tko + \Delta Tres + Af.\ (F/Fo)^{0.33} \quad (2)$$

As we mentioned above there are new data supporting the *model for "lateral (horizontal) shift"* of the critical transition temperature curve after neutron re-irradiation [1, 2, 3]. According to this model the re-embrittlement rate significantly decreases in comparison to the conservative law:

$$Tkf = Tko + (\Delta Tres^3 + A f^3 . (F/Fo))^{0.33} \qquad (3)$$

According Russian standard for weld metal with P content up to 0.04% an Δ Tres = 20°C is accepted. This value is correct for annealed Unit 2 and Unit 3, but for Unit 1 the P content is significantly higher. For this case a value of 40°C is proposed in [3]. The Δ Tres values are shone in table 5.

Table 5

	Unit 1	Unit 2	Unit 3	Unit 4
Δ Tres , °C	40	20	20	-

3. RPV radiation life time assessment

The contemporary values of P and Cu concentrations, Tko and Δ Tres are used for prediction of the neutron embrittlement of the weld metal . The trend curves are calculated using designed fluence (G) and the calculated mean values (CAL)[6,7] at 1/4 wall thickness (D). The results obtained by "conservative" (CON) and " horizontal" (LAT) re-embrittlement calculation model for Tkf are compered on figures 1,2 and 3.

The designed maximal allowed Tka values are used for RPV radiation life time prediction.

The life time extension of RPV metal for Unit 1,2,3 and 4, determinated by different methods are compared in table 6. The new method for RPV life time determination by maximal allowed flaw is proposed by Gidropress. This method is applied now on Unit 1 and designed life time (30 years) is proved in the case of validity of conservative re-embrittlement law.

Table 6

EOL		Unit 1			Unit 2			Unit 3			Unit 4
[years]											
	CON		LAT		CON	LAT	CON		LAT		
F	G	CAL	G	CAL	G	G	G	CAL	G	CAL	G
[1/4 D]											
Tka											
[°C]											
163	1992	1993	1997	1999	2002	2009	1999	2003	2006	2011	2119
193	1996	1998	2004	2009	2014	2021	-	-	-	-	-
210	-	-	-	-	-	-	2019	2032	2021	2035	228

4. Conclusions

- In the case of validity of the lateral re-embrittlement law for P content up to 0.05% all RPV reach or exceed their designed life time;
- The installation of MSIV is obligatory for each of the first three Units;
- After MSIV installation Unit 2 and 3 reach their designed life time;
- For Unit 4 (low impurity content and low Tko) no problems connected with neutron embrittlement of RPV are expected;
- The use of the real fluence values in Tkf calculation results in a radiation life time extension;

- A new verification of the chemical composition of RPV Unit 1 weld 4 metal is recommended;
- A standardisation of new Gidropress life time determination by maximal allowed flaw and determination of the ultra sonic inspection limit flaw resolution are necessary;
- The fracture mechanics methods and thermo-shock hydraulic conditions should be re-assessed and developed, so that more accurate determination of the maximal permissible critical temperature (Tka) could be achieved. Only in this case it could be possible to obtain an exact prediction of the residual life time.

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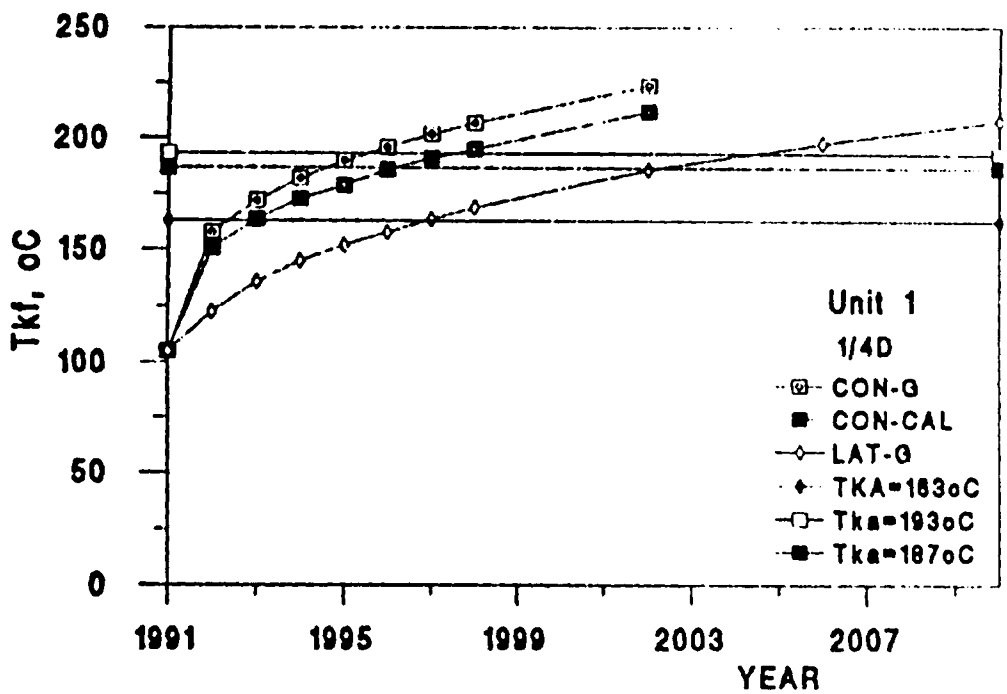

Fig. 1

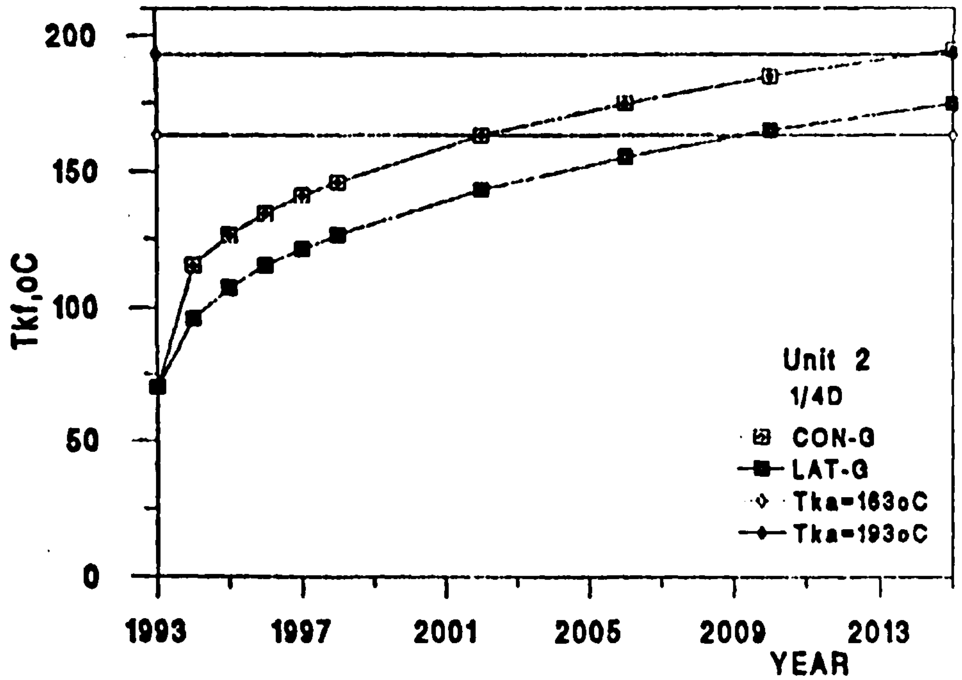

Fig. 2

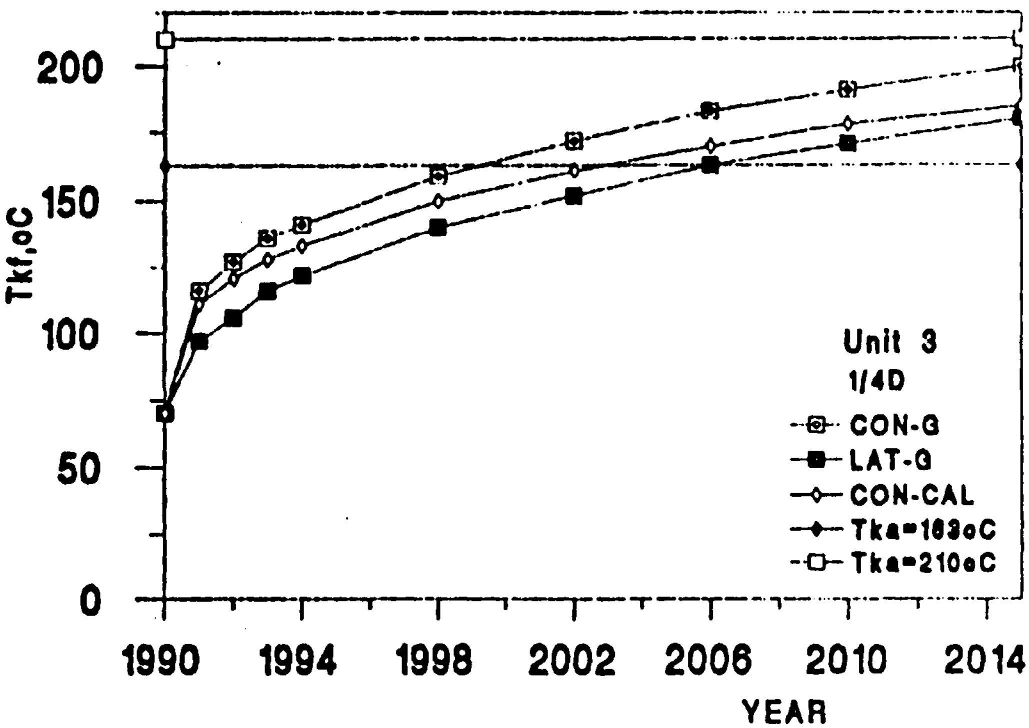

Fig. 3

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NEUTRON FLUENCE DETERMINATION FOR AN ASSESSMENT OF OPERATION EFFECTIVENESS AND PREDICTION OF THE VVERs PRESSURE VESSEL LIFE TIME AT KOZLODUY NPP

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One of the most important nuclear safety problems is the materials quality assurance. The factors affecting the degradation and ageing of reactor pressure vessel (RPV) material are:

- the metal quality;
- the processing manner in the pressure vessel manufacturing;
- radiation environment;
- chemical environment;
- thermal conditions.

Besides the counted up factors also is possible occurring of unexpected synergy effects from the mentioned above factors.

During the recent years intensive investigations on the embrittlement processes have been carried out and the knowledge for metal ageing has been continuously in progress.

The study of fast neutrons effect on the materials, expressed by the neutron fluence, requires maximum accurate determination of neutron flux and spectrum, which, on its hand, has set up the ***Pressure Vessel Reactor Dosimetry.***

The critical temperature Tk, which is a quantitative measure for the metal radiation brittleness is empirically related to the neutron fluence F by:

$$Tk = Tk(F)$$

The neutron fluence is determined:

- at the places experiencing the most severe irradiation influence on the base metal and the welds at opposite of reactor core:
  . on the inner pressure vessel wall
  . on the outer pressure vessel wall
- in the surveillances with sample-witnesses and detectors.

An assessment of the critical temperature at the internal wall can be done by nondestructive analysis only using the calculated fluence.

By the neutron fluence are determined:

• the current embrittlement;

• the operation efficiency;

• the RPV life time.

The development of our knowledge for the radiation embrittlement of VVER-440 and VVER-1000 pressure vessels has been based on:

- utilization of ready made solutions and instructions of the VVER Manufacturer;
- accumulation of self made experience during the operation;
- outcomes from the accumulated experience;
- collaboration with leading institutions.

The accumulated experience shows that it is not sufficient to utilize the Manufacturer's ready made solutions, but it is also necessary to develop self experience in solving the radiation embrittlement problems.

The main activities in gaining self experience for the radiation embrittlement assessment of **VVER-440** pressure vessels are shown below:

1985: • acquisition with the problem, actuality of its solution for the reactors at NPP Kozloduy;

1987: • development of a code package for fluence calculations;

1989: • development of methodology for determination of the neutron fluence and the critical temperature of radiation embrittlement based on a two-dimensional geometry model;

• calculational evaluation was accomplished for the fluence and the critical temperature of unit 3;

1991: • calculational evaluation was accomplished for the fluence and the critical temperature of unit 2;

The achieved experience and obtained results had encouraged the Kozloduy NPP Director at that time eng. Z. Boyadzhiev to take a *nontraditional* for the plant decision. Opposing the Manufacturer's instruction for shutting down the unit, an assessment of neutron fluence due to the real reactor operation was proposed and carried out. The obtained results justified the decision that Unit 2 would work out the rest 6 months of the 16th cycle.

An initial Surveillance programme for the VVER-440 PV was established jointly with the Kozloduy NPP specialists;

• the activities on the Surveillance programme for the VVER-440 PV metal were presented and confirmed by the OSART specialists.

1992: • calculational evaluation was made of the neutron fluence, the critical temperature and the life time of Units 3 and 4;

1993: • further development of the code package by incorporating a three-dimensional geometry model;

• an assessment was performed for the neutron fluence, the critical temperature of radiation embrittlement and the life time of Unit 1 at Kozloduy NPP;

• chemical analysis of weld 4 Unit 1 was made;

• detectors were installed behind the pressure vessels of Units 1,2,3 and 4 according to the reactors worldwide safety demands;

1994: • a final Surveillance programme was developed and accepted for the PV metal of VVER-440;

• Unit 1 - comparative analysis of shavings activities, sampled after the 14th cycle was performed for verification of the calculated fluence.

The accumulation of self experience for the **VVER-1000** radiation embrittlement assessment began in

1991: • development of calculation model

• proposal for an initial Surveillance programme for the VVER-1000 PV metal;

1992: • testing of the three dimensional geometry model;

1993: • evaluation of the neutron fluence and the radiation state of Unit 5 PV for the operated cycles 1 to 3;

1994: • evaluation of the neutron fluence and the radiation state of Unit 6 PV for the operated cycle 1;

• installation of activation detectors behind the pressure vessel of Unit 5 and 6;

• a Surveillance programme was developed and finally accepted for the pressure vessel metal of VVER-1000.

The following *main conclusions* could be made *on the base of the accumulated experience:*

- a staff methodology is necessary to be developed for continuous neutron fluence monitoring by utilization of the elaborated and applied by us digital reactivity meter DR-8;

- a correlation methodology is necessary to be developed between the data obtained from the experimental tests of the surveillances (staff sample-witnesses) and the fluence at some critical places at the pressure vessel.

The developed experience and methodology have been enlarged and improved in *collaboration with the leading worldwide institutions like:*

1985 - SCODA, pressure vessels manufacturer;

1986 - Institute for Nuclear Investigations, Rez, the critical assembly LR-0; experiments simulating VVER-440 and VVER-1000;

1987 - The Kurchatov Institute - the project constructor;

1988 - Finland, calculations of neutron fluence and measurements by detectors;

1990 - KORPUS - a reactor system for mock-up experiments, RIFSCR „RIAR" Dimitrovgrad, Russia;

1991 - EWGRD&WGRD-VVER - European Workshop on Reactor Dosimetry (Belgium, Czech Republic, Russia, Germany, France, Finland, USA, Slovakia, Hungary, Italy, Bulgaria) annual working meetings and participation in the ASTM - conferences.

All these activities have been fulfilled with the active contribution from the Kozloduy NPP specialists: P. Tzokov, J. Gledachev, Iv. Ivanov, EP-1 - N. Nelov and group, V. Tzocheva - sector Rodiochemy, Tz. Haralampieva and group, A. Zlateva and group, EP-2 - V. Velichkov, M. Monev, T. Batachka, Vl. Jovchev.

RESULTS:

Block-chart of the calculation procedure for neutron fluence calculation in the most common manner is represented on Fig. 1.

The accumulation of neutron fluence dependant on the real operation of Unit 1 (Fig. 2) shows that the requirements for fluence restriction are not satisfied even after the unit annealing. Additional studies and analyses are necessary for determination of the metal characteristics after annealing.

The results for neutron fluence of unit 2 (Fig. 3) show that the safety operation demands determine the radiation resource up to cycle 31 in case of applying dummy cassettes.

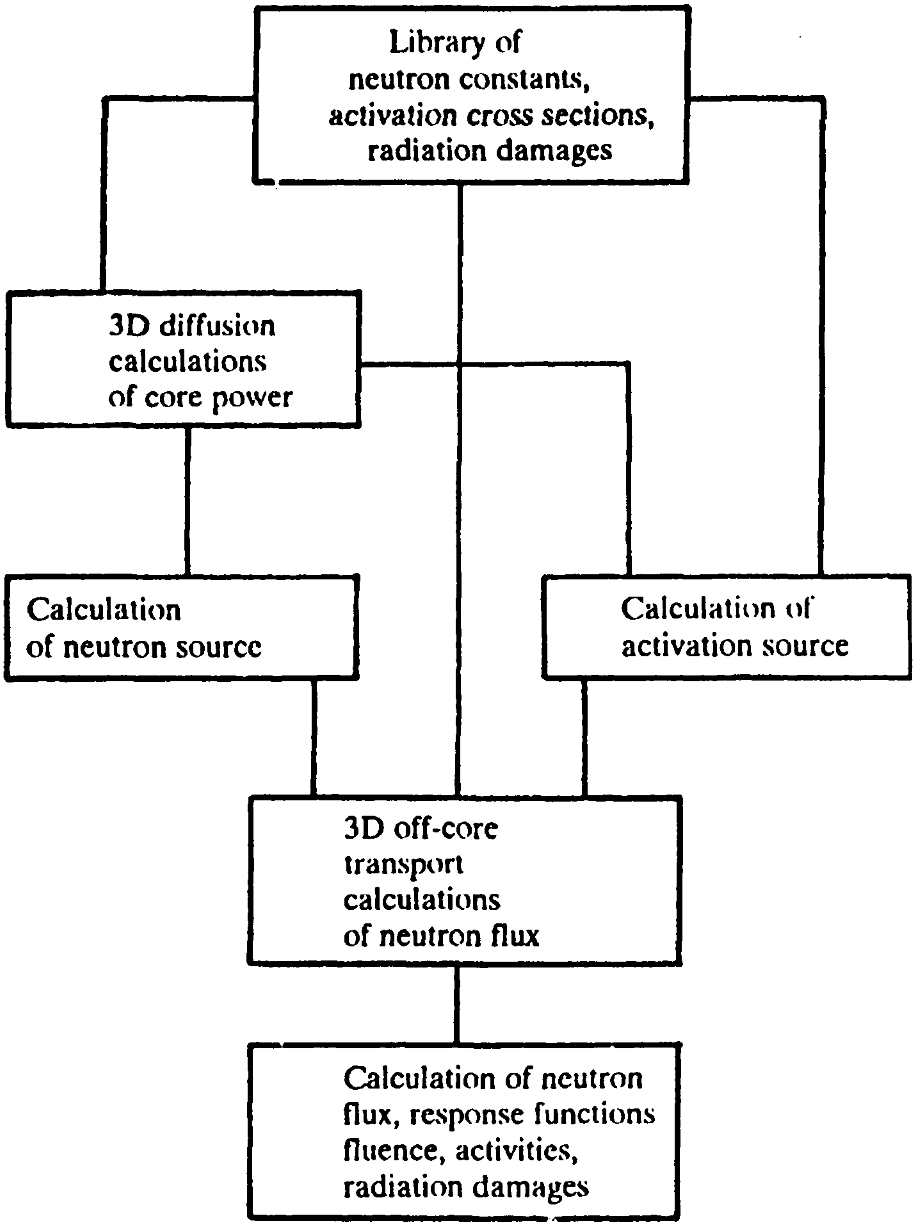


Fig.1 Neutron fluence calculational methods

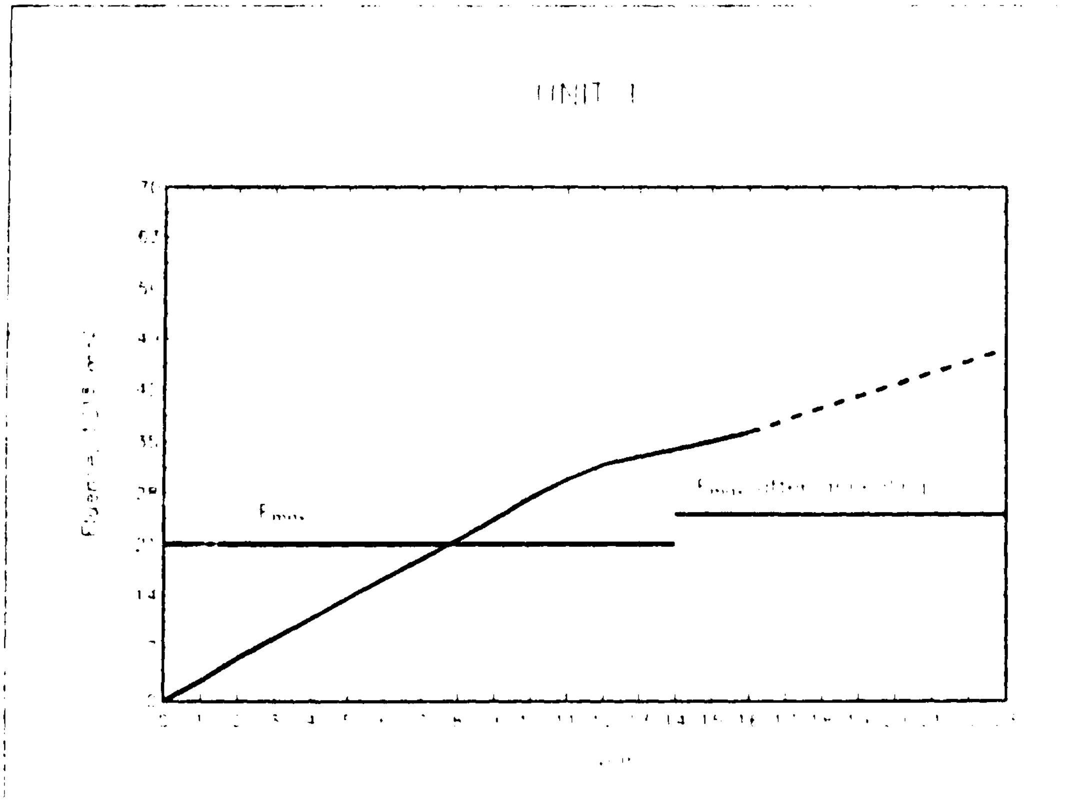


Fig. 2 Accumulation of neutron fluence dependant on the real operation and prognoses

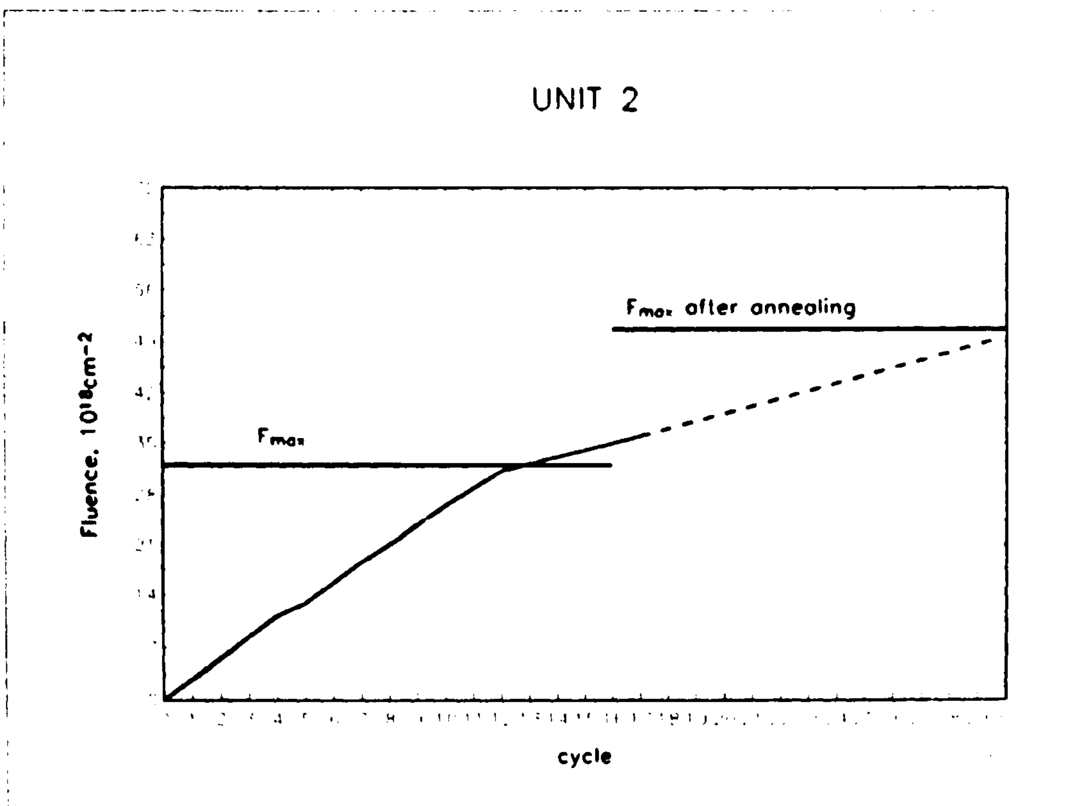


Fig. 3 Accumulation of neutron fluence dependant on the real operation and prognoses

As it can be seen from Fig. 4, Unit 3 can be safely operated up to the end of the project life time (30 years), but only under a working regime with dummy cassettes.

UNIT 3

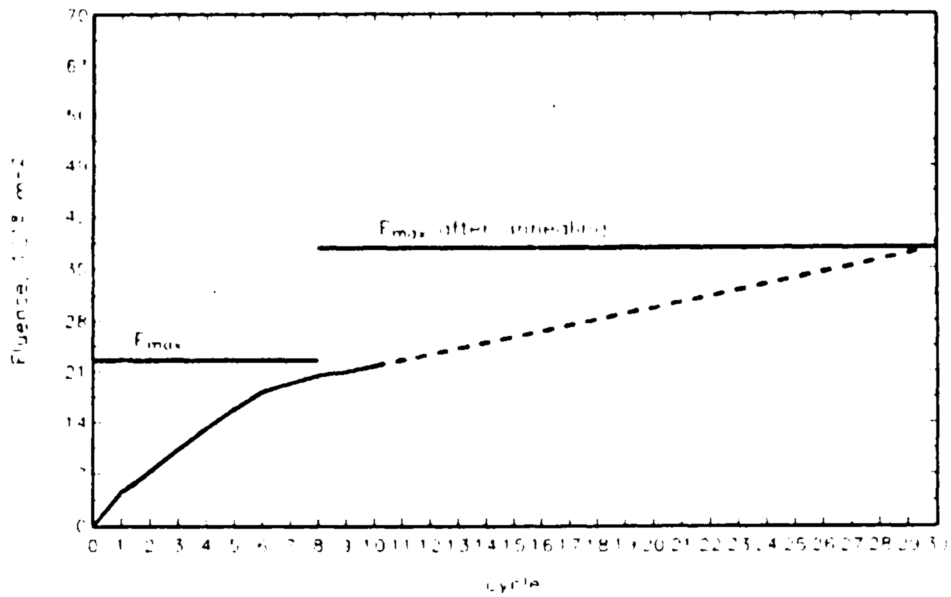


Fig. 4 Accumulation of neutron fluence dependant on the real operation and prognoses

UNIT 4

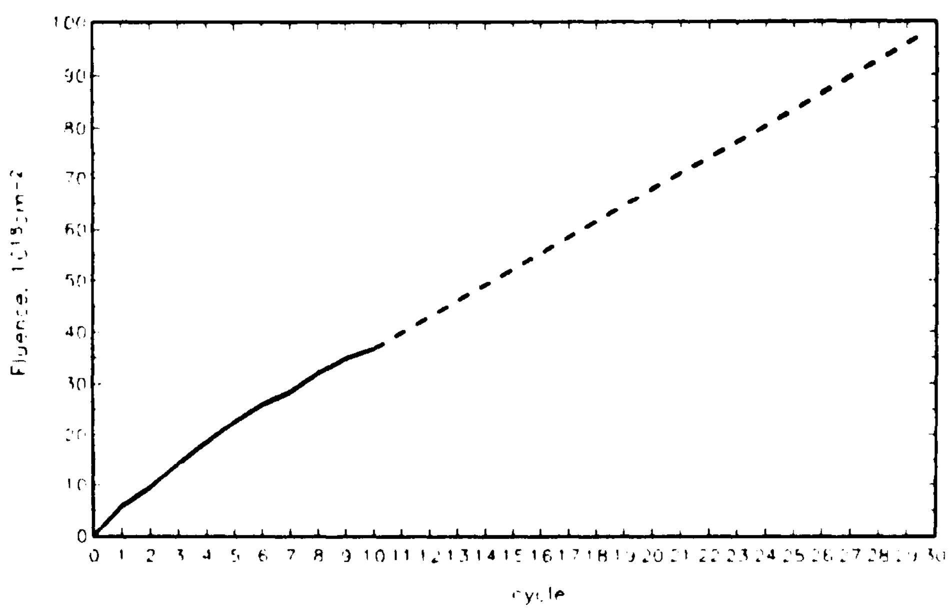


Fig. 5 Accumulation of neutron fluence dependant on the real operation and prognoses

The pressure vessel of Unit 4 (Fig. 5) is in the most satisfactory state from the view point of the metal radiation embrittlement. In the condition that all the data handed out by the Manufacturer are not revised the reactor could be effectively operated with standard loadings up to the end of its project lifetime.

An assessment of the temperature of metal radiation brittleness has been made on the basis of the calculated fluence for the pressure vessels of Units 1 to 4 (Fig. 6-9).

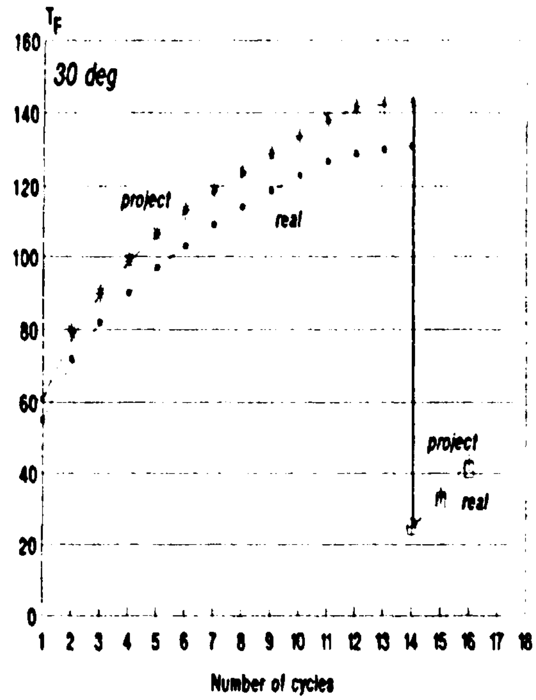


Fig. 6 Shift of the temperature of metal radiation brittleness, Unit 1, Kozloduy NPP

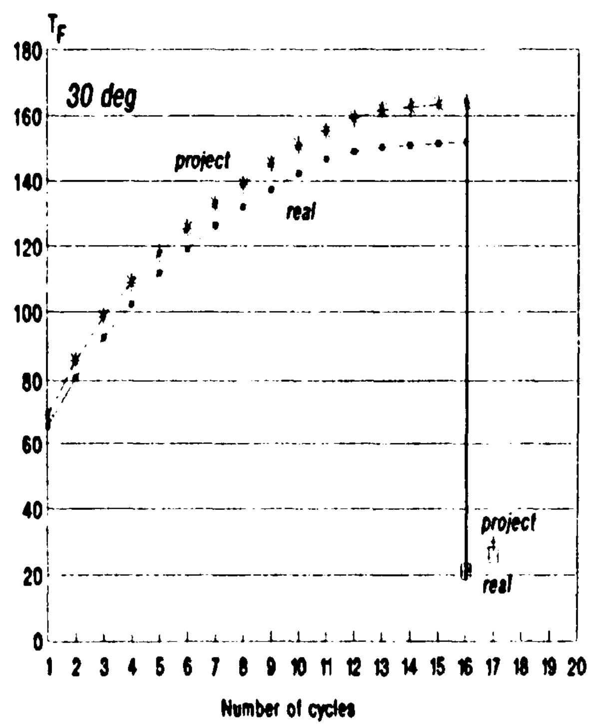


Fig. 7 Shift of the temperature of metal radiation brittleness, Unit 2, Kozloduy NPP

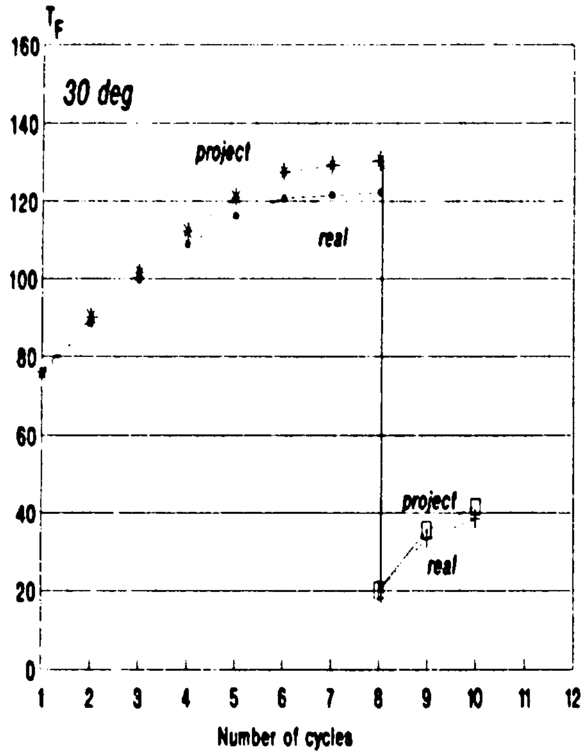


Fig. 8 Shift of the temperature of metal radiation brittleness, Unit 3, Kozloduy NPP

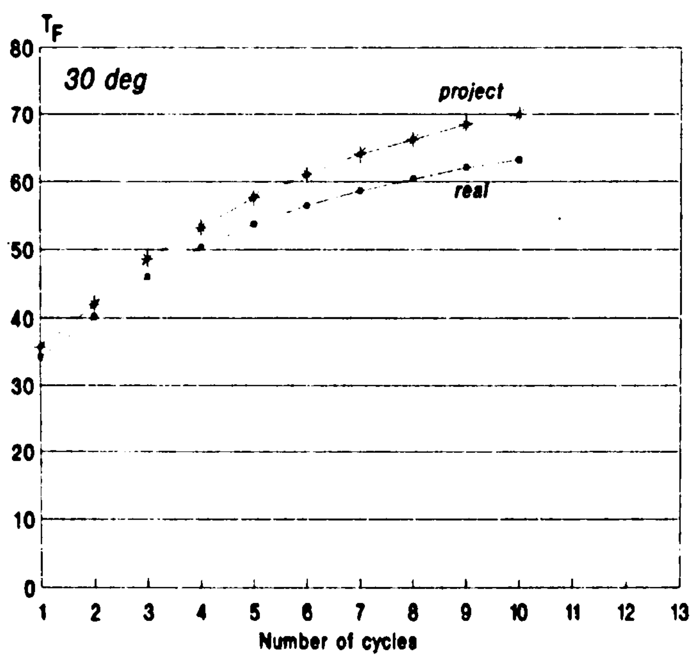


Fig. 9 Shift of the temperature of metal radiation brittleness, Unit 4, Kozloduy NPP

On Fig.10 the change in the critical temperature of embrittlement is represented for the operated 17 cycles of unit 2 and some forecast assessments accounting for the "conservative" and "lateral" law of shifting the critical temperature after annealing. Coming out from the analysis of the obtained results is that:

- applying the "conservative" law and in average projected annual augment of the neutron fluence $F=1.62\times10^{18}cm^{-2}$ the reactor could be operated safely up to the 22 cycle;

- applying the "lateral" law and in average (experienced in the unit real operation) annual augment of the neutron fluence $F=1.2\times10^{18}cm^{-2}$ a reliable operation of the unit will be ensured up to the 31 cycle and in $F=1.62\times10^{18}cm^{-2}$ - up to the 27 cycle.

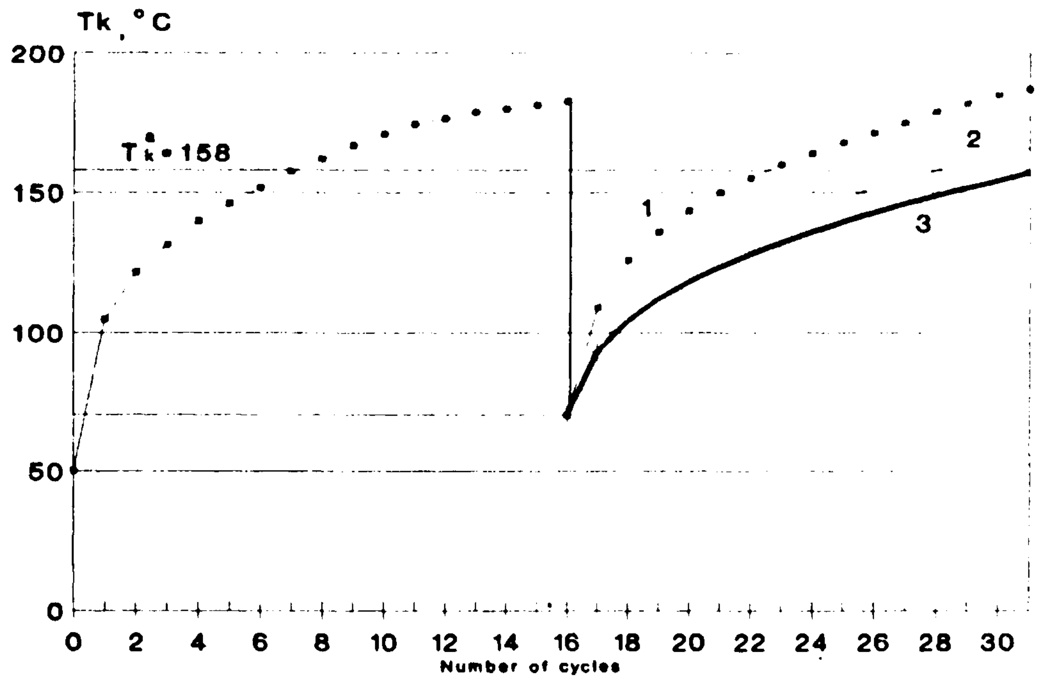


Fig.10 Shift of the critical brittlenes temperature Tĸ and prognoses, Unit 2, NPP Kozloduy, line 1 - „conservative" law, $F=1.62\times10^{18}cm^{-2}$; line 2 - „lateral" law, $F=1.62\times10^{18}cm^{-2}$, line 3 - „lateral" law, $F=1.2\times10^{18}cm^{-2}$

The accumulation of fluence onto the VVER-1000 pressure vessel, Unit 5 is shown in Fig. 11. The tendency that the fluence, obtained from the real operation is lower than the project one is a guarantee for the unit safe operation.

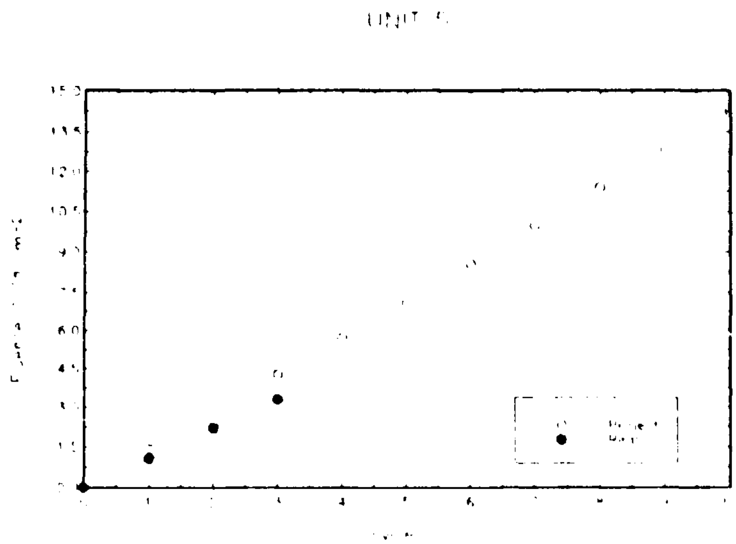


Fig. 11 Accumulation of neutron fluence dependant on the real operation and prognoses

Measurements and calculations have been carried out for determination of the activities of shavings taken out from the internal wall of weld 4 after the 14th cycle of Unit 1, Kozloduy NPP. These measurements provide an unique opportunity for appropriate verification of the neutron fluence calculations.

Table 1. The measured and calculated activities of the shavings referred 01.01.1993.

Place	Depth	Activity, E+5 Bq/g	
N^0	mm	Measured	Calculated
(2)-0^0	3.4-4.1	1.34 ± 0.04	1.14
(2)-0^0	4.1-4.8	1.32 ± 0.04	0.984
(3)-0^0	4.0-4.8	1.25 ± 0.04	0.984

CONCLUSIONS:

A methodology has been developed for neutron fluence determination with respect to the contemporary high-tech requirement.

The accurately determined fluence can be used for:

- a choice of reloading schemes with maximum reduced fluence;
- an assessment of the current state of the pressure vessel metal;
- an assessment of the operation effectiveness;
- a prediction of the rest life time.

It is necessary to be developed jointly with EP-2:

- a staff methodology for continuous neutron fluence monitoring;

- a data base for the neutron exposure and damages to be used for retrospective analyses.

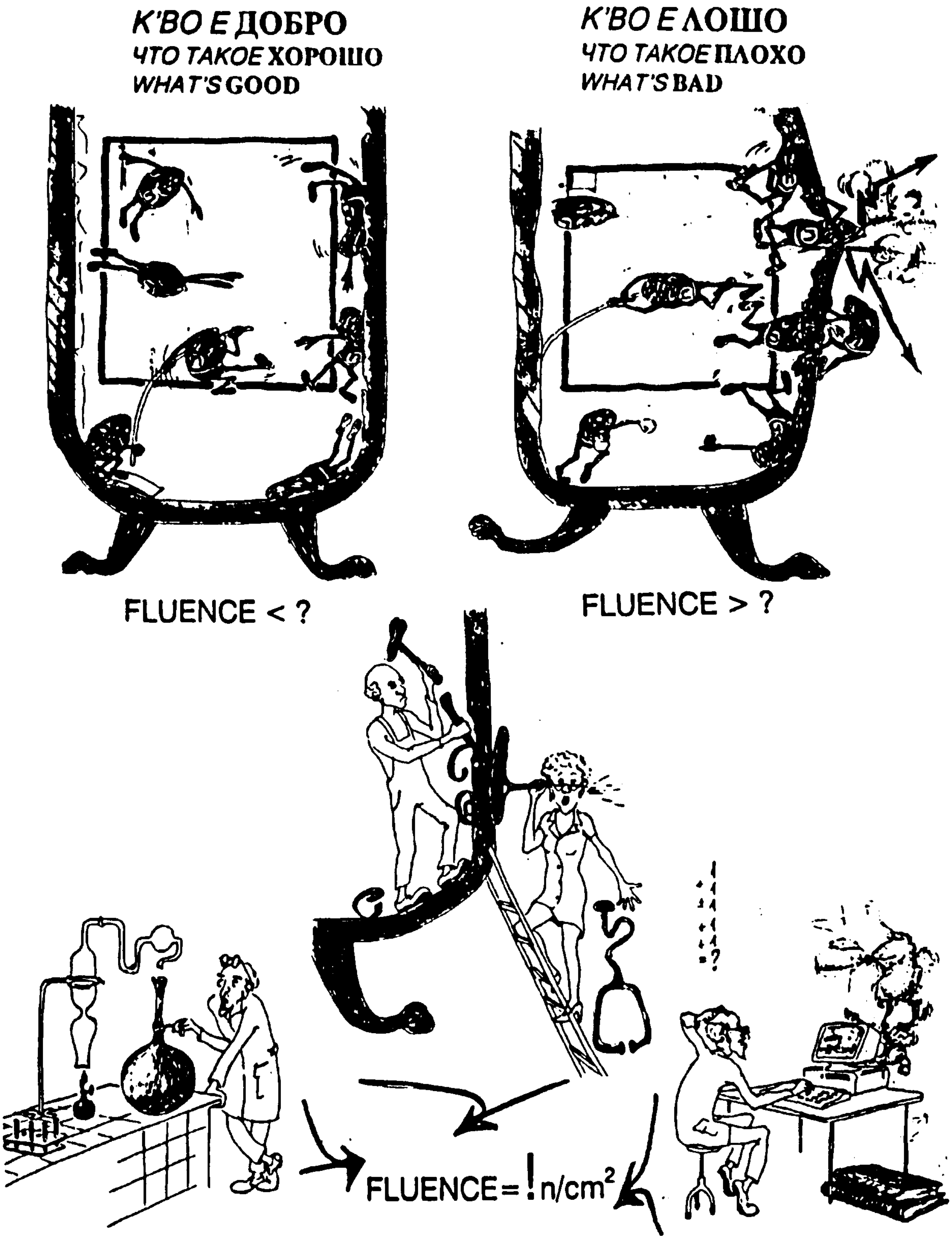
For the purposes of quality assurance are necessary:

- ex-vessel activity measurements; simulating experiments at the LR-0 critical assembly, Rez,

Benchmark comparative analyses;

- enhancement of the reliability on the base of self and distinct achievements.

НЕУТРОННО ОБЛЪЧВАНЕ НА РЕАКТОРНИЯ КОРПУС



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NUCLEAR FUEL UTILIZATION IN KOZLODUY NPP

dipl. eng. Z. Boyadjiev, M. Sc.Ts. Haralampieva, Ts. Peychinov

Fuel utilisation in Kozloduy NPP as a matter of principle strives after efficiency and economy with safe reactor operation criteria met.

In compliance with this principle the core loadings of units 1-6 are designed so that at optimum quantity of fresh fuel and fuel cycle duration the basic operation parameters should be within the admissible limits , providing for nuclear safety. For this purpose in Kozloduy NPP the following reasearch and developments are being carried out in terms of:

- extended reactor physics calculations and analyses;
- improving the accuracy of the computer codes used to model and calculate the processes occurring in the core;
- study the fuel behaviour under operation conditions;
- evaluation of the fuel assemlies condition by applying the methods for checking fuel cladding integrity at the end of each fuel cycle
- actions to reduce the irradiation of the reactor pressure vessels;
- improvement and development of the in-core monitoring systems;
- working-out of an optimized programme for refuellings and outages,etc.;

Up to the present moment the nuclear power reactors in Kozloduy NPP have been operated for a total of 62 fuel cycles.

Units 1 and 2 utilizing VVER-440/B-230 reactors, which in 1991 were shut down for upgrading and modernization, are presently running their 17th and 18th fuel cycles. Tables 1 and 2 show some basic characteristics of core loading and operation conditions during past fuel cycles, for Unit 1 and Unit 2, respectively. Tabulated are:

- fuel enrichment and number of fresh fuel assemblies;
- fuel cycles duration in effective days;
- maximum fuel power peaking factors at design calculations Kqcalc and during reactor operation at Kqmeas, taken for one and the same moments from the fuel cycles;
- reactor capacity factor;
- average discharge burn-up in MWD/kgU;

The tables show the basic variations of the core loading design /1/ approved by the respective cognizant authorities and the fuel supplier:

- core design change principles amounting to placing more heavily burned-up assemblies in the periphery of the core, i.e. low-leakage core loading pattern;
- deployment of 36 dummy assemblies in the core periphery, protecting the reactor pressure vessel from the damaging impact of the fast neuton flux.

The average design value of the discharge fuel burn-up has been reached (31 MWd/kgU).

Unit 3 was commissioned in 1981. It has been operated for 12 fuel cycles, the characteristics of which are given in Table 3. Also tabulated is the specific burn-up of U^{235} in the fresh fuel, representing the ratio of U^{235} mass to the number of effective days in the design fuel cycle duration. Since the 7th fuel cycle, aiming at reduction of the reactor pressure vessel irradiation, thirty-six dummy assemblies have been placed in the core in combination with a low-leakage core loading pattern. The rest of the core periphery has been supplied with 24 or 30 fuel assemlies with considerable burn-up. (Fig.1)

Since the 10th fuel cycle, the enrichment of a number of the fuel portions of the control assemblies have been increased from 2.4% of U^{235} (design) to 3.6%.

The core loading selection method adopted since the 10th fuel cycle includes the following actions:

- to observe, if possible, core loading pattern by alternating 90,90,96 fresh fuel assemblies of 3.6% enrichment in the course of 3 successive fuel cycles;

- to utilize 12 fresh control assemblies of 3.6% enrichment in every cycle and every second year 1 control assembly of 2.4% enrichment in central position;

- extension, if necessary, of the design duration of the fuel cycles utilizing a power effect.

By meeting the above mentioned conditions provision is made for 300 effective days of the fuel cycles compared to 260-270 by design if dummies are utilized. The average discharge fuel ourn-up is approximately 33-34 MWD/kgU.

Unit 4 of Kozloduy NPP was commissioned in 1982. It has got an improved technical design compared to that of Units 1 and 2 and to a greater extent meets the raised safety requirements to nuclear power units. Up to the present moment the reactor has completed 11 fuel cycles. The number of assemblies in the core corresponds to design - 349, of them 312 fuel assemblies and 37 control elements.

Table 4 shows the basic core loading and operation characteristics over the past 11 fuel cycles.

Since the 4th fuel cycle low-leakage core loading patterns have been used by placing more heavily burned-up assemblies in the peripherey of the core. At the same time, part of the fresh fuel assemblies are moved to the centre of the core. (Fig.2)

The adoption of the low-leakage core loading patterns increases the fuel cycles duration by approximately 10% compared to design, while for steady-state fuel cycles the increase is 3-4%, the amount of fresh fuel being the same.

Since the 7th fuel cycle of Unit 4 the core loading design has been based on the following fundamental principles:

- forty-eight assemlies of considerable burn-up are deployed in the core periphery;

- part of these periphery assemblies have been used in the core for 3 fuel cycles and their integrity has been checked;

- gradually adopt the practice of utilizing control assemblies the fuel portion of which is enriched to 3.6%;

- the height of the working group of control assemblies in the core with the reactor at power is 200 cm;

- it is possible to use a power effect to extend the fuel cycle in compliance with grid demand and nuclear safety principles.

As can be seen from Table 4, by means of the core loading described above the average duration of the fuel cycles achieved under operation at nominal parameters is 300-310 effective days and the average number of fresh fuel assemblies (both working and control) is about 102-108 compared to design 114-120.

The average discharge fuel burn-up achieved is within 33-36.5% MWD/kgU.

The average burn-up of 3.6% enrichment control assemblies discharged upon completion of their 3rd fuel cycle is within 32-35.7 MWD/kgU. Since 1986 / Unit4's 6th fuel cycle / core loading designs are performed by means of computations using the SPPS-1 code.

The multiple comparisons made on the basis of operational data from Loviisa NPP and Kozloduy NPP show that the code is precise and adequate to compute both low-leakage core loading patterns and cores utilizing dummies. Fig.3 shows the basic core parameters corresponding to the operating conditions of Unit 3 during its 10th fuel cycle.

Plotted are also the dependence of boron critical concentrations and the fuel burn-up obtained from computations and they have been compared to the operational data. Results show that in rated power reactor operation modelling the measured and computed boron concentrations values concur very well.

To obtain the measured relative power Kq^{meas} the data from the regular temperature control system have been processed according to the procedure adopted in the plant /4/. Since units 1-4 have not got an automatic data acquisition and processing system yet, these have been used without being statistically processed.

Tables 5, 6 and Fig.4 show the basic results of the computed Kq^{calc} and the measured Kq^{meas} peaking power factors for Unit 3' and Unit 4'. A lot of summarised results are also shown concerning the errors (deviations) of the measurement and the computation. The mean square errors are within 2-3 % and they can be higher only when the reactor is operated at a power level lower than rated. The maximum positive and negative deviations are within +7%.

The maximum measured values of the power peaking factors in fuel are predicted by computation with precision better than 4%.

The systematic analyses conducted on the power distribution in the cores of Units 1-4 show that it is possible to ensure efficient operation without violation of the peaking factors permissible limits.

We came to the conclusion, however, that the accuracy of the computation is commensurate with the operation measurements accuracy. The forthcoming implementation of the automatic parameter monitoring systems for the WWER-440 reactor cores in Kozloduy NPP will intensify the reactor physics analyses and will provide a better systemizaton of operational data.

Units 5 and 6 of Kozloduy NPP utilize WWER-1000 reactors of the third generation which have incorporated higher techical-economical indicators. Compared to the WWER-440 they have increased fuel power density and core sizes, the electric power has also been increased to 1000 MW, the mechanical reactivity control system makes use of cluster assemblies, the in-core monitoring system has been improved.

For the first core loadings of WWER-1000 a two-year fuel cycle has been applied, the fuel having an average fuel enrichment level of 3.3% and average discharge burn-up 28.5 MWD/kgU. The same was applied for Unit 5 and 6 reactors.

Unit 5 was put into service in November 1987. At present it has been running its 4th fuel cycle. The basic operational characteristics of the core loadings of Unit 5 are shown in Table 7. For 4th fuel cycle the computed data for the cycle life time and the average burn-up are given. The first loading was carried out according to design. At the end of the 2nd fuel cycle a power effect was used in the course of 33 effective days. For the 3rd fuel cycle (Fig.5) a low-leakage core loading pattern has been partially applied and provision was made for a negative temperature effect at the beginning of the cycle by changing the places of 6th and 8th group of the control cluster assemblies.

The reactor of Unit 6 reached criticality level on 29th May 1991. At present it has been running its 2nd fuel cycle (Fig.6). The basic core loading characteristics of Unit 6 are shown in Table 8.

The core loading option of WWER-1000 (Unit 5 and 6) are chosen by means of the computer code developed by Kurchatove Nuclear Research Institute, Russia (BIPR-7) /5/ and by VNIIAES (ALBOM, PROROC) /6,7/

As can be seen on Tables 8 and 9, the fuel cycles of Units 5 and 6 do not exceed the admissible peaking factors values and the burn-up values.

Table 9 and 10 show the results of comparisons made between computed and operation values of the relative fuel power distributions for some moments of the fuel cycles. The operation values are obtained by processing the current of the sensors located in the neutron measurement channels. The VMPO system, performing this processing, shows greater errors for these fuel assemblies that have no neutron measurement channels. The maximum error in the shown comparisons apply to such assemblies. The errors in fuel assemblies having maximum energy release do not exceed 6%.

The three-year fuel cycle proved to be more efficient for WWER - 1000. The average enrichment level in this type of cycle is 4.4% and the average computed burn-up - 43 MWD/kgU. It is foreseen Units 5 and 6 also to adopt the 3-year fuel cycle. The latter is more economical and allows better utilization of fuel. The fuel cost within the energy cost is reduced approximately by 15% compared to that relative of the 2-year cycle. The

Research and Development department of Kozloduy NPP - 2 carried out computations and analyses to provide the basis for the adoption of the 3-year fuel cycle.

CONCLUSION

The given results enable us to make some conclusions and suggestions.

1. The core loading option chosen for WWERs in Kozloduy NPP (Units 1-6) has led to efficient utilization without violation of nuclear safety criteria.

2. By utilizing dummies and low-leakage core-loading patterns for WWER-440 cores (Units 1-4) we can reach effective reduction of the reactor presure vessel irradiation. The increased enrichment level of the fuel portion of the control rods and the improved characteristics of Unit 3 and 4 fuel cycles have lead to maintaining the design duration of the fuel cycles at a reduced number of assemblies by a factor 5-10%

3. The improvement of the in-core monitoring system for WWERs-440 (B-230), the implementation of on-line simulation of the processes occurring in the core together with the intended implementation of new more accurate computer codes will enables us to utilize fuel with higher enrichment and implement the 4-year fuel cycle.

4. For the WWERs-1000 the adoption of the 3-year fuel cycle utilizing fuel of 4.4% initial enrichment will ensure better core loading characteristics and more effective fuel utilization.

5. The improvement of the in-core monitoring system for the WWER-1000 will allow to better the accuracy of the operation data and will considerably increase the reliability of nuclear fuel control. This will create conditions to implement improved fuel with a new type of absorbers and the more effective low-leakage core loading patterns.

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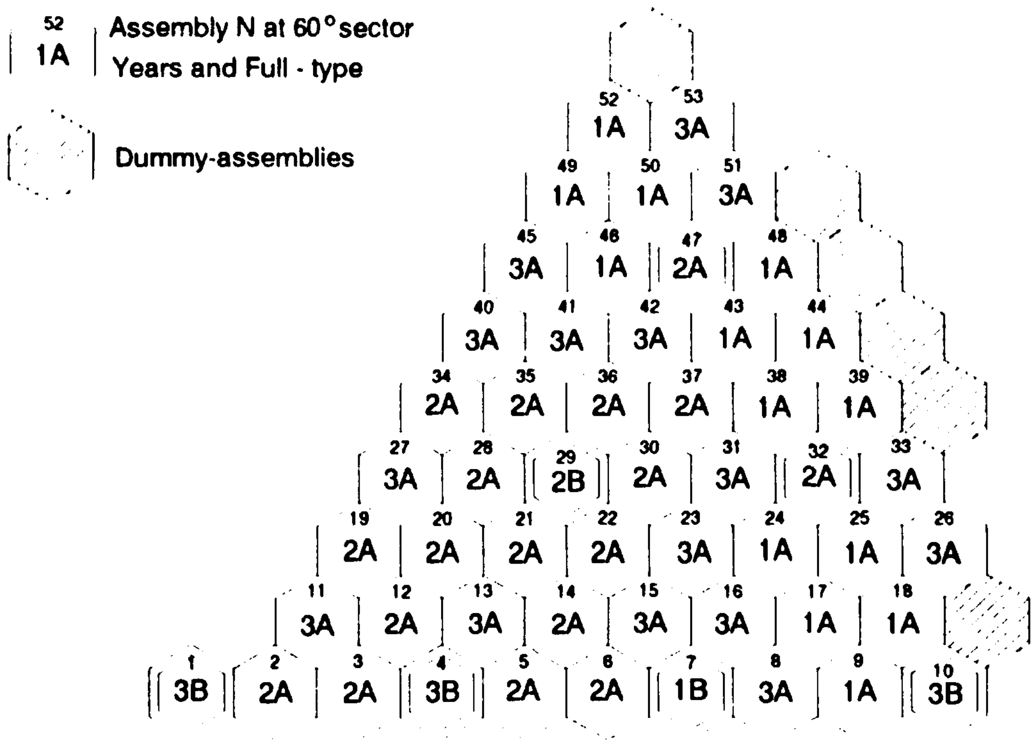


Fig.1. Unit 3 Cycle 11

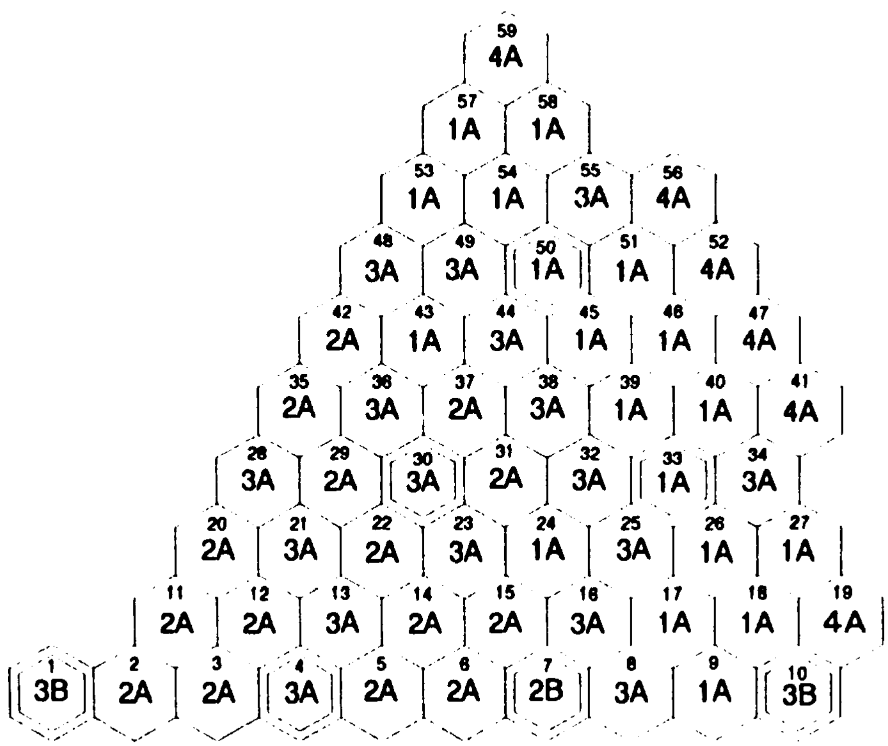
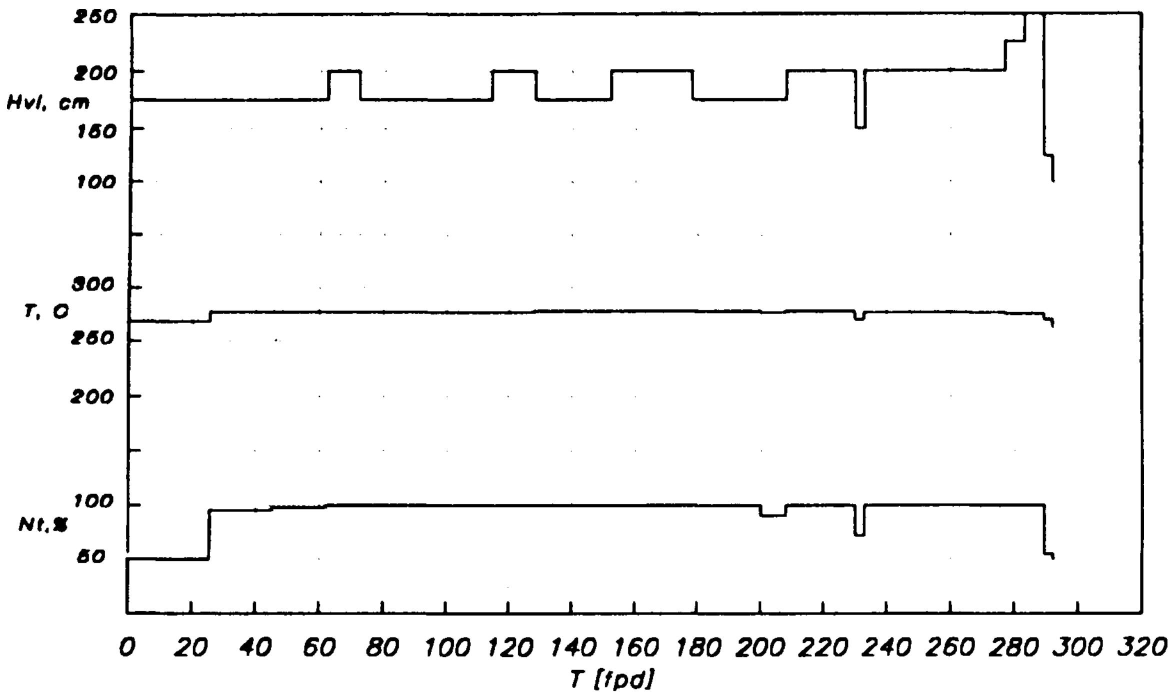


Fig.2. Unit 4 Cycle 12



Reactor thermal power, coolant average temperature and requlating control rod group position as a functions of energy production in full power days during 10 cycle, Unit 3.

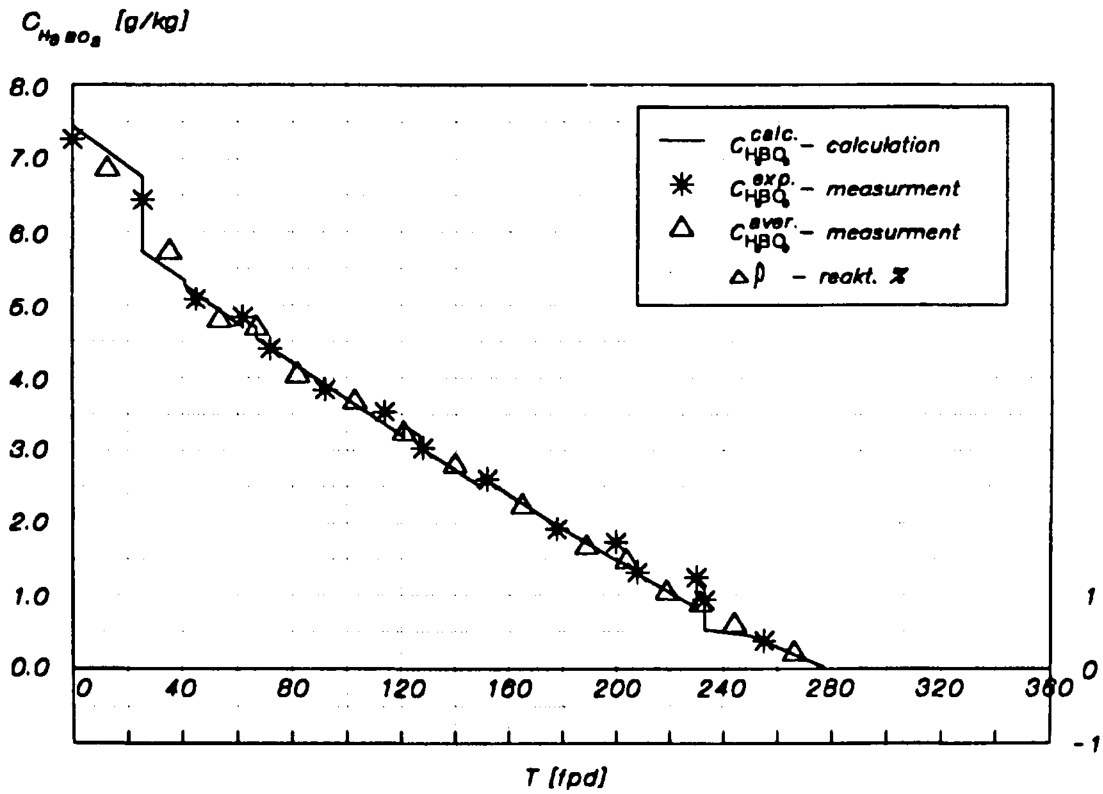
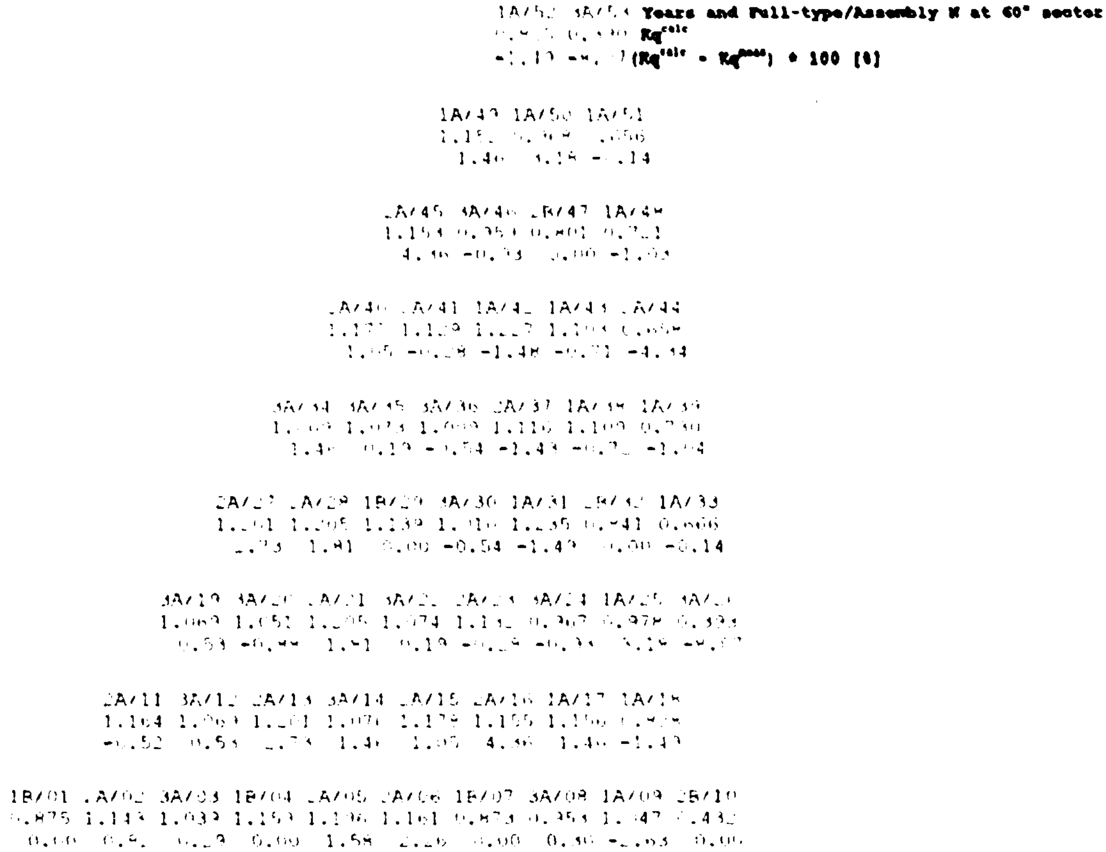


Fig. 3 Boric acid concentration changes (calculated and measured) during10 cycle, Unit 3.



Keff= 0.99516 Power=-100.0 Tin= 262.7 CB= 0.542 H6=175.00 G= 35000.0
Max + 4.4/ 16/A2 Disp = 2.4 Max Kq^{meas} & Kq^{calc}= 1.250 1.235/ 31
Max - -8.1/ 26/A3 E-ARK= 0.1 Max Kq^{calc} & Kq^{meas}= 1.235 1.250/ 31
E-ster= 2.0 E-Rad= -2.9 E-Rad 1 2 3 = -1.5 -1.3 -8.1
Err & Disp by Fuel-type and Years
A1: 90.0x -0.2 A2: 96.0x 1.1 A3: 90.0x -1.0 A4: 0.0x 0.0

Figure 4 Distributions of assemblywise relative power Kq^{calc}, calculated by SPPS-1 and absolute error from measurments (Kq^{calc} - Kq^{meas}).100 [%] (Cycle 7, Unit 3, FPD=152)

POOR QUALITY ORIGINAL

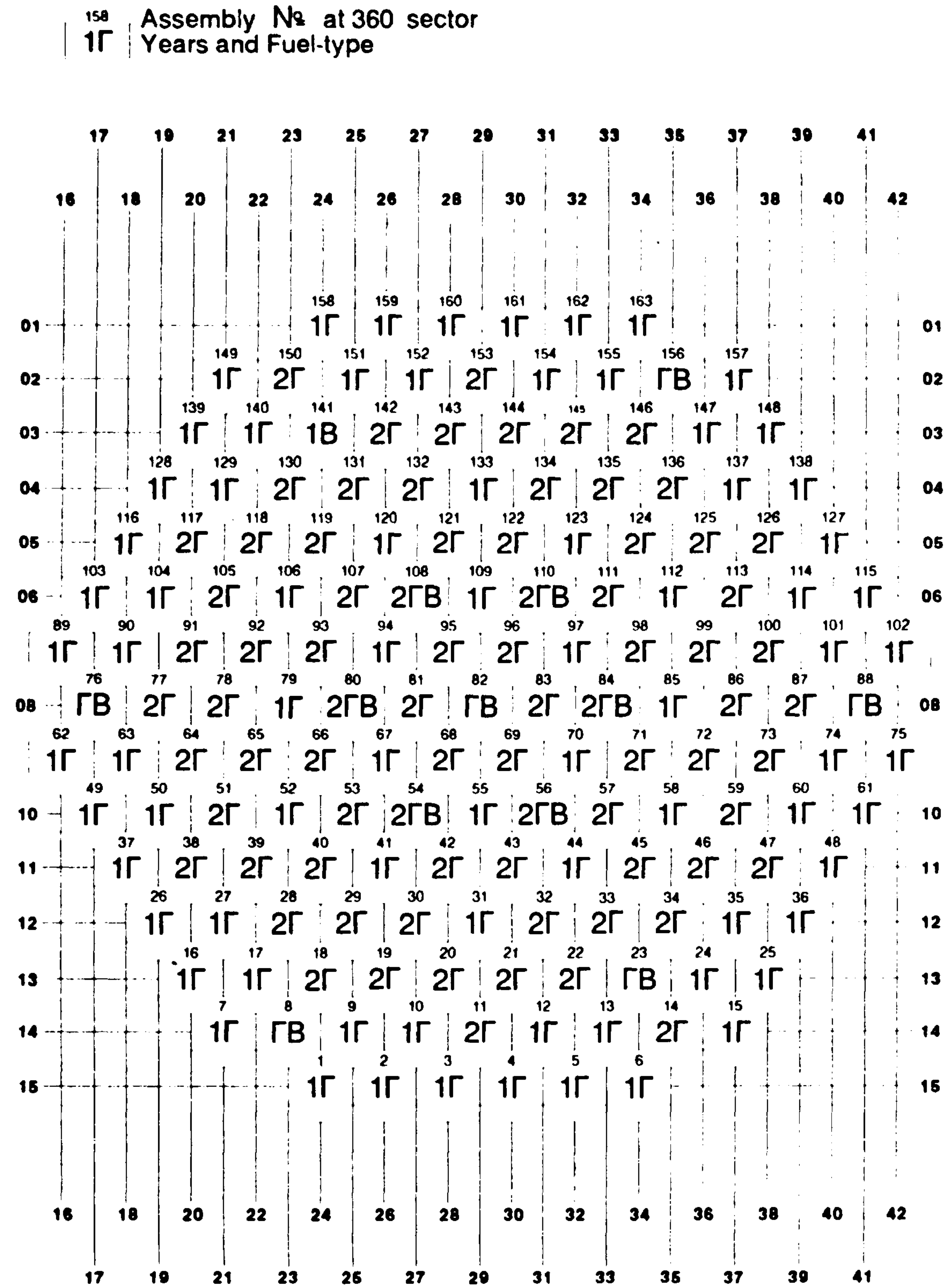


Fig.5 Unit 5 Cycle 3

A - enrichment 2.0% Г - enrichment 3.3%
B - enrichment 3.0% ГВ - enrichment 3.3% prof. with 3.0%

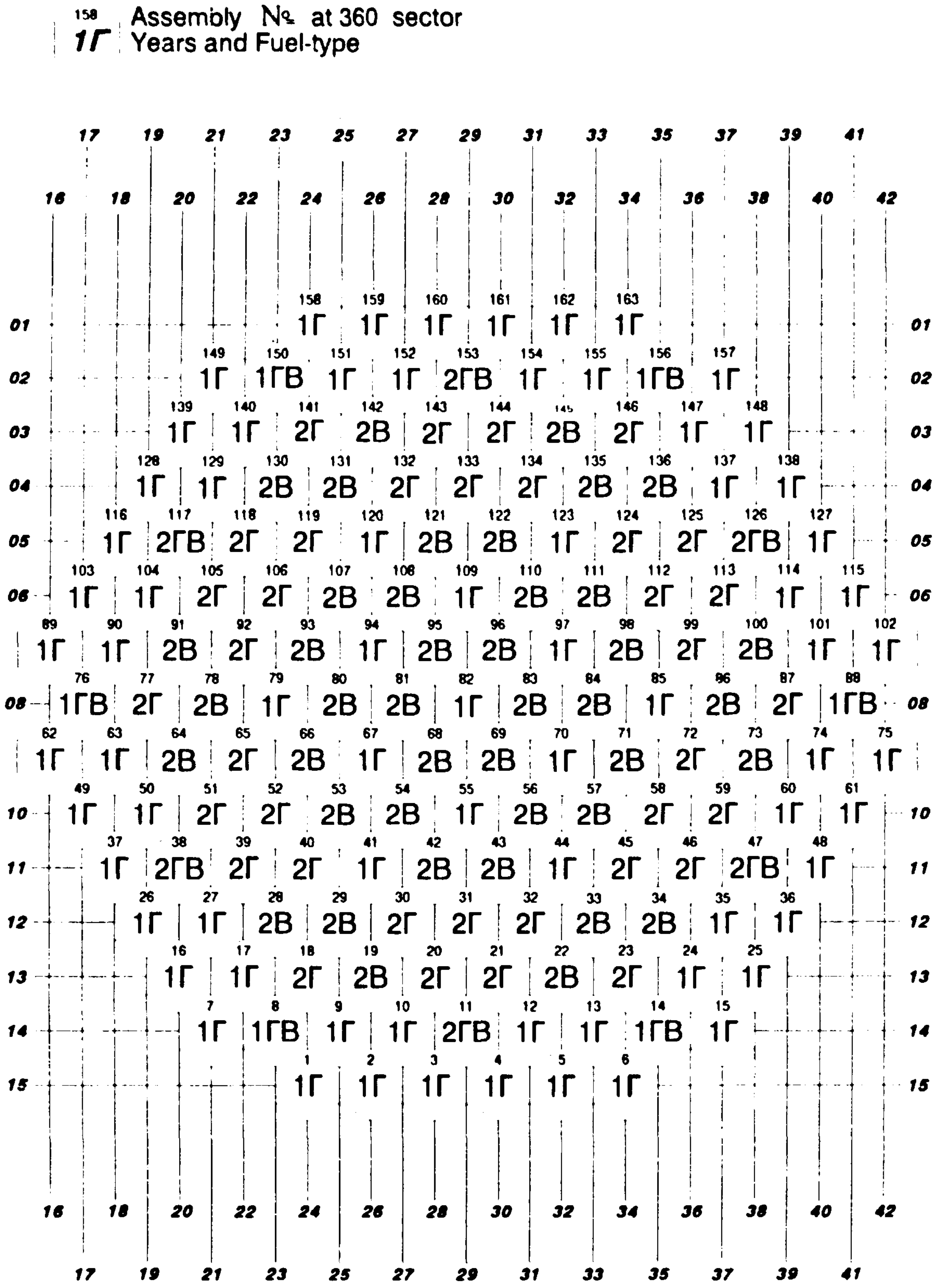


Fig.6 Unit 6 Cicle 2

Table 1

Cycle No	Loaded assemblies: ACA enrichment	ACA number	FWA enrichment	FWA number	Cycle life time FPD	TPAF %	Spec consumption of U^{235}	calc Kq /FPD	meas Kq /FPD	Core periphery spec features	Average burnup of withdrawn assemblies [MWd/kgU] 1.6%	2.4%	3.6%
1	1B 1C	25 12	1A 1B 1C	102 108 102	287.79	0.59		1.300	1.31	1A-60 1B- 6	9.33	8.73	
2	1B 1C	12 25	1A	102	357.40	0.77		1.324 /27	1.34 /30	1A-60 1C- 6	10.24	22.3	
3	1B	13	1A 1B	102 6	303.60	0.87	1.61	1.255 /7	1.27 /16	1A-60 2B- 6	19.13		32.98
4	1B	12	1A	102	311.70	0.95	1.50	1.268 /2	1.26 /12	1A-60 3B- 6		23.53	32.37
5	1B 1C	13 1	1A 1B	102 6	317.70	0.92	1.54	1.311 /3	1.24 /2	1A-60 2B- 5 1C- 1		28.22	32.01
6	1B	12	1A 1B 1C	96 3 3	308.30	0.95	1.48	1.310 /2	1.29 /6	1A-60 1B- 6	15.58	27.62	31.68
7	1B	13	1A 1B 1C	90 6 6	297.65	0.92	1.51	1.275 /3	1.28 /3	1A-60 1B- 6	19.16	27.97	31.16
8	1B	12	1A 1B 1C	72 36 5	302.70	0.95	1.50	1.278 /3	1.292 /6	1A-60 1B- 6	18.09	26.34	30.70
9	1B	13	1A	102	312.87	0.93	1.51	1.272 /5		1A-60 1B- 6	19.04	27.08	30.68
10	1B	13	1A 1B	102 6	329.23	0.94	1.48	1.23 /25	1.27 /28	1A-60 2B- 1 3B -5		29.16	32.29
11	1B	12	1A	90	310.30	0.95	1.35	1.255 /8	1.35 /7	1A-36 2A-12 4A-12 2B- 6		27.61	32.67
12	1B	13	1A 1C	90 1	293.70	0.95	1.44	1.278 /1	1.31 /1	1A-24 3A-12 4A-24 3B- 6	10.24	28.76	31.56
13	1B	12	1A	84	270.60	0.96	1.45	1.266 /14	1.27 /14	KE-36 1A-12 4A-12 3B- 6		24.23	32.80
14	1B	18	1A	90	304.50	0.95	1.43	1.245	1.286	KE-36 3A-12 4A-12 3B- 6		23.10	31.97
15	1B	7	1A 1B	84 3	277.00	0.92	1.40	1.225 /16.5	1.220 /16.5	KE-36 1A-12 3A- 1 4A- 11 3B- 6		26.63	33.79
16	1B	12	1A 1B	96 6	274.00	0.85	1.68	1.228 /28	1.256 /28	KE-36 1A-24 3B- 6		27.55	32.31

Basic data on the previous 16 fuel cycles, carried out on Unit 1

ACA - Automatic Control Assemblies
WFA - Working Fuel Assemblies
TPAF - Thermal Power Availability Factor
FPD - Full Power Days

A - enrichment 3.6%
B - enrichment 2.4%
C - enrichment 1.6%

Table 2

Cycle N	Loaded assemblies: ACA enrichment	ACA number	FWA enrichment	FWA number	Cycle life time FPD	TPAF %	Spec consumption of U^{235}	calc Kq /FPD	meas Kq /FPD	Core periphery spec. features	Average burnup of withdrawn assemblies [MWd/kgU] 1.6%	2.4%	3.6%
1	1A 1C	25 12	1A 1B 1C	102 108 102	290.2	0.82		1.268 /15	1.260 /15	1A-60 1B- 6	9.57		
2	1B	12	1A	72	284.8	0.89		1.237 /40	1.290 /40	1A-60 2B- 6	17.83	20.34	
3	1B	12	1A 1B	96 42	345	0.99	1.63	1.228 /5	1.252 /5	1A-60 1B- 6		26.14	31.37
4	1B	13	1A	102	312.2	0.92	1.51	1.237 /2	1.264 /2	1A-60 1B- 6		29.19	31.68
5	1B	12	1A 1B	66 6	2[illegible]3.5	0.98	1.56	1.278 /5	1.287 /5	1A-60 2B- 6		28.10	30.77
6	1B	13	1A 1B	84 6	290.4	0.97	1.42	1.262 /2.7		1A-60 1B- 6		23.76	32.79
7	1B	12	1A 1B	90 12	328.7	0.88	1.38	1.240 /8.0	1.282 /8.0	1A-60 2B- 6		26.39	33.04
8	1B	13	1A 1B	90 6	278	0.94	1.58	1.197 /6.4	1.203 /6.4	1A-60 1B- 6		26.81	33.01
9	1B	12	1A 1B	84 6	298.9	0.92	1.37	1.258 /8.7	1.250 /8.7	1A-60 2B- 6		28.40	32.41
10	1B	13	1A 1B	90 14	339.3	0.95	1.36	1.269 /3.0	1.269 /3	1A-36 4A-24 3B- 6		25.47	32.10
11	1B	12	1A	102	310.5	0.94	1.51	1.283 /2	1.313 /2	1A-36 4A-24 2B- 6		25.52	32.73
12	1B	18	1A	96	322	0.91	1.43	1.301 /10	1.277 /10	1A-24 3A-12 4A-23 3B- 7		21.89	34.27
13	1B	19	1A	90	320.2	0.79	1.37	1.320 /8	1.340 /8	1A-12 3A-24 4A-24 2B- 6		21.23	32.48
14	1B	12	1A 1B	90 4	303.1	0.91	1.42	1.261 /15	1.29 /15	KE-36 1A-12 3A- 6 4A- 6 3B- 6		23.52	34.82
15	1B	6	1A	90	280.6	0.92	1.43	1.270 /4	1.250 /4	KE-36 1A-12 3A- 6 4A- 6 3B- 6		28.30	34.29
16	1B	19	1A 1B	90 2	300	0.87	1.48	1.282 /7.8	1.270 /7.8	KE-36 1A-12 3A-11 4A- 1 3B- 6		27.65	31.75
17	1B	12	1A 1B	96 6	293.4	0.73	1.57	1.236 /12.4	1.23 /12.4	KE-36 1A-12 3A- 6 4A- 4 2B- 2 3B- 6		26.87	33.72

Basic data on the previous 17 fuel cycles, carried out on Unit 2

ACA	- Automatic Control Assemblies	A	- enrichment 3.6%
WFA	- Working Fuel Assemblies	B	- enrichment 2.4%
TPAF	- Thermal Power Availability Factor	C	- enrichment 1.6%
FPD	- Full Power Days		

Table 3

Cycle N	Loaded assemblies: ACA enrichment	ACA number	FWA enrichment	FWA number	Cycle life time FPD	TPAF %	Spec consumption of U^{235}	calc Kq /FPD	meas Kq /FPD	Core periphery spec. features	Average burnup of withdrawn assemblies [MWd/kgU] 1.6%	2.4%	3.6%
1	1C 1B	12 25	1A 1B 1C	102 108 102	406	82		1.340 /2	1.364 /2	1A-60 1B- 6	13.3	-	
2	1B	12	1A 1B	102 6	270.2	92		1.262 /9	1.211 /9	1A-60 2B- 6	-	23.8	
3	1B	13	1A 1B	84 12	289.8	96	1.482	1.240 /14	1.266 /14	1A-60 1B- 6	-	29.1	32.7
4	1B	12	1A 1B	84 7	330.9	97	1.246	1.265 /7	1.28 /7	1A-36 4A-24 2B- 6	-	26.2	31.4
5	1B	13	1A	102	335.8	95	1.406	-	-	1A-36 4A-24 2B- 6	-	26.6	33.8
6	1B	18	1A	96	293.8	86	1.568	1.278 /58	1.296 /58	1A-24 3A-12 4A-24 3B- 6	-	21.9	33.1
7	1B	19	1A	90	300.5	95	1.457	1.258 /5	1.270 /5	1A-36 2A- 6 3A-12 2B- 6	-	19.5	32.8
8	1B	12	1A	90	326.4	96	1.281	1.265 /5	1.282 /5	1A-24 3A-30 3B- 6	-	22.1	32.9
9	1B	19	1A	96	322.5	95	1.437	1.264 /5	1.280 /5	1A-30 3A-24 3B- 6		24.0	34.8
10	1B 1A	6 12	1A	96	292.1	88	1.636	1.289 /26	1.280 /26	1A-30 3A-24 2B -6	-	24.7	35.1
11	1B	6	1A	84	310.4	82	1.209	1.259 /9	1.242 /9	1A-30 3A-24 3B- 6	-	26.0	34.1
12	1B 1A	7 12	1A 1B	90 7	330.5	69	1.437	1.275 /6	1.270 /6	1A-24 3A-33 4B- 3	-	-	34.7

Basic data on the previous 12 fuel cycles, carried out on Unit 3

ACA	- Automatic Control Assemblies	A	- enrichment 3.6%
WFA	- Working Fuel Assemblies	B	- enrichment 2.4%
TPAF	- Thermal Power Availability Factor	C	- enrichment 1.6%
FPD	- Full Power Days		

Table 4

Cycle N	Loaded assemblies ACA enrichment	ACA number	FWA enrichment	FWA number	Cycle life time FPD	TPAF %	Spec consumption of U^{235}	calc Kq /FPD	meas Kq /FPD	Core periphery spec. features	Average burnup of withdrawn assemblies [MWd/kgU] 1.6%	2.4%	3.6%
1	1B 1C	25 12	1A 1B 1C	102 108 102	390.0	0.90		1.276 /11.4	1.298 /11.4	1A-60 1B- 6	12.8		
2	1B	12	1A	102	241.2	0.95		1.271 /16.8	1.300 /16.8	1A-60 2B - 6		22.2	
3	1B	13	1A 1B	90 6	348.9	0.97	1.26	1.358 /7.5	1.346 /7.5	1A-36 2A-12 1B - 6 3B -12		26.5	33.2
4	1B	12	1A	102	313.7	0.96	1.50	1.361 /9	1.270 /9	1A-36 2A-12 4A-12 2B - 6		25.6	32.7
5	1B	19	1A	96	354.4	0.96	1.31	1.306 /3.9	1.240 /3.9	1A-24 3A-12 4A-24 3B- 6		21.3	33.9
6	1B	18	1A	96	309.5	0.96	1.49	1.307 /19	1.301 /19	1A-12 3A-30 4A-18 2B - 6		21.1	34.1
7	1B 1A	7	1A	90	244.0	0.90	1.66	1.316 /20.9	1.288 /20.9	1A-12 3A-18 4A-30 2B - 6		20.3	33.2
8	1B	6	1A	78	350.4	0.91	1.00	1.292 /19	1.301 /19	1A-12 4A-48 3B - 6		26.4	34.4
9	1A	12	1C 1B 1A	1 6 90	295.0	0.95	1.54	1.338 /2.7	1.339 /2.7	1A-12 4A-48 3B - 6		25.8	35.1
10	1B 1A	7 12	1A	90	309.3	0.87	1.47	1.335 /3.8	1.270 /3.8	1A-12 4A-48 3B - 6	13.9	24.2	36.1
11	1B	6	1A	90	332.0	0.64	1.21	1.326 /5.4	1.310 /5.4	1A-12 3A-12 4A-36 2B - 6		29.7	36.5
12	1A	12	1A	96	297.4		1.49	1.337		1A-12 3A-12 4A-36 3B - 6		21.7	34.4

Basic data on the previous 12 fuel cycles, carried out on Unit 4

ACA	- Automatic Control Assemblies	A	- enrichment 3.6%
WFA	- Working Fuel Assemblies	B	- enrichment 2.4%
TPAF	- Thermal Power Availability Factor	C	- enrichment 1.6%
FPD	- Full Power Days		

Table 5

T, FPD / Parameters / of conditions / in tde core /	3.5 / H_{VI}=197cm / T_{in}=264.2°C / N_T = 90 % / C_B=1.007g/kg	5.4 / H_{VI}=193cm / T_{in}=263°C / N_T = 100% / C_B= 0.963g/kg	209.9/ H_{VI}=200cm / T_{in}=262.3°C / N_T = 100% / C_B=0.180g/kg
Keff	1.0018	1.0020	0.9970
\Keff - calcul error [%]	0.0018	0.0020	-0.0030
Max. K_q			
K_q^{meas}/K_q^{calc} (Nass)	1.260 / 1.267 (22)	1.253 / 1.260 (22)	1.212 /1.208 (51)
K_q^{calc}/K_q^{meas} (N_{ass})	1.267/1.260 (22)	1.260/1.253 (22)	1.213/1.166 (228)
Maximum positive error	6.0	7.3	4.7
N of assembly (type)	82 (2A)	82 (2A)	228 (2A)
Maximum negative error	- 6.4	- 5.2	- 3.1
N of assembly (type)	1 (3A)	20 (1A)	230 (2A)
σ mean-square error	2.7	2.7	1.8
mean error 1A (84)	0.1	- 0.4	1.1
mean error 2A (96)	1.8	2.1	0.4
mean error 3A (95)	- 1.9	- 1.7	- 1.4
error peripherial assemblies	-2.4	-2.4	-0.2

Main results in SPPS - 1 calculation, compared with the measured assemblywise power distributions Kq(n) for different moments [FPD] of the Cycle 11 Unit 3
Error Kq(n) = $(K_{q(n)}^{calc} - K_{q(n)}^{meas}) \cdot 100$ [%] n - location number in 360° sector

Table 6

T, FPD / Parameters / of conditions / in tde core /	5.44 / H_{VI}=181cm / T_{in}=262.1°C / N_T = 100% / C_B=1.06 g/kg	33.5 / H_{VI}=191cm / T_{in}=261.7°C / N_T = 100% / C_B=0.91 g/kg	67.5 / H_{VI}=193cm / T_{in}=262.4°C / N_T = 100% / C_B=0.79 g/kg	275 / H_{VI}=180cm / T_{in}=264.4°C / N_T = 100% / C_B=0.068 g/kg
Keff	0.9989	1.0011	0.9995	0.9952
\Keff - calcul error [%]	-0.11	0.11	-0.05	-0.48
Max. K_q				
K_q^{meas}/K_q^{calc} (Nass)	1.314/1.327 (48)	1.315/1.318 (305)	1.304/1.312 (305)	1.295/1.276 (305)
K_q^{calc}/K_q^{meas} (N_{ass})	1.327/1.314 (48)	1.320/1.315 (48)	1.314/1.304 (48)	1.277/1.295 (142)
Maximum positive error	4.8	6.2	5.4	5.8
N of assembly (type)	52 /1A/	72 /4A/	90 /4A/	91 /1A/
Maximum negative error	- 3.6	- 6.1	- 4.7	- 6.8
N of assembly (type)	78 /3A/	257 /3A/	272 /3A/	272 /3A/
σ mean-square error	1.8	2.5	2.3	2.8
mean error 1A (78)	0.7	0.0	0.4	1.4
mean error 2A (90)	0.5	0.1	0.0	0.5
mean error 3A (96)	-1.1	-1.7	-1.8	-2.7
mean error 4A (48)	-1.3	3.6	3.4	2.4
mean error 3B (6)	-1.3	-0.5	-0.9	-3.4
error peripherial assemblies	0.6	3.1	3.0	1.6

Main results in SPPS - 1 calculation, compared with the measured assemblywise power distributions Kq(n) for different moments [FPD] of the Cycle 11 Unit 4 Error Kq(n) = $(K_{q(n)}^{calc} - K_{q(n)}^{meas}) \cdot 100$ [%]
n - location number in 360° sector

Table 7

Cycle	Loaded assemblies type	Loaded assemblies number	Cycle life time FPD	crit. C_{H3BO3} g/kg	Relative power distribution factors BOC Kq	BOC Kv	EOC Kq	EOC Kv	Average burnup of withdrawn assemblies [MWd/kgU]
1.	A B Г ГB	79 42 36 6	298.6	6.53	1.31	1.81	1.20	1.35	12.670 15.059
2.*	A B Г ГB	- - 73 6	325.0	6.95	1.25	1.56	1.20	1.33	23.786 28.180 24.742 26.806
3.	B Г ГB	1 78 6	317.7	7.77	1.25	1.63	1.18	1.31	28.180 24.742 26.806
4.	B Г ГB	- 72 6	308.6	7.49	1.27	1.62	1.20	1.33	27.170 27.040 24.540

Main cycle characteristics, Unit 5

A - enrichment 2.0%
B - enrichment 3.0%
Г - enrichment 3.3%
ГB - enrichment 3.3% prof. with 3.0%
* - stretch-out

Table 8

Cycle	Loaded assemblies type	Loaded assemblies number	Cycle life time FPD	crit C_{H3BO3} g/kg	Relative power distribution factors BOC Kq	BOC Kv	EOC Kq	EOC Kv	Average burnup of withdrawn assemblies [MWd/kgU]
1.	A B Г ГB	79 42 36 6	301.6	6.53	1.31	1.81	1.20	1.35	12.865 15.360
2.	B Г ГB	- 73 6	298.2	7.13	1.29	1.65	1.19	1.31	27.280 23.505 26.230

Main cycle characteristics, Unit 6

A - enrichment 2.0%
B - enrichment 3.0%
Г - enrichment 3.3%
ГB - enrichment 3.3% prof. with 3.0%

Table 9

Parameters of conditions in the core	T, FPD			
	75	153	208	275
H_{10}, cm	280	273	280	278
Tin, °C	283.2	285.5	285.2	283.5
NT, %	79	94	95	77
C_{H3BO3}, meas., g/kg	6.01	3.72	2.54	1.24
C_{H3BO3}, calc, g/kg	6.00	3.88	2.54	1.32
meas calc Kq_{max}\Kq (Nass)	1.26/1.23 60	1.23/1.20 129	1.19/1.19 129	1.19/1.18 10
calc meas Kq_{max}\Kq (Nass)	1.23/ 1.26 60	1.20/1.23 129	1.19/1.19 129	1.18/1.19 10
Max. (+) difference N of assembly (type)	8.70 123 (Г)	9.74 44 (Г)	6.80 1 (Г)	7.33 123 (Г)
Max. (-) difference N of assembliy (type)	5.43 11 (Г)	4.63 153 (Г)	5.83 14 (Г)	4.57 11 (Г)
All assemblies root mean square	2.79	2.80	2.30	2.89
Assemblies with SPD root mean square	1.43	1.57	2.54	2.97
Assemblies without SPD root mean square	3.10	3.11	2.11	2.74

Main results in BIPR-7 calculation, compared with the measured assemblywise power distributions Kq(n) for different moments [FPD] of the Cycle 3 Unit 5

$$\text{Difference} = \frac{Kq_{(n)}^{calc} - Kq_{(n)}^{meas}}{Kq_{(n)}^{meas}} \cdot 100\,[\%]$$

n - location number in 360° sector

Table 10

Parameters of conditions in the core	T, FPD			
	24	42.5	123.8	221.7
H_{10}, cm	279	278	274	268
Tin, °C	287.5	286.9	285.1	282.5
NT, %	99	93	79	51
C_{H3BO3}, meas., g/kg	6.14	5.82	4.22	2.42
C_{H3BO3}, calc, g/kg	6.32	5.97	4.02	2.51
meas calc Kq_{max}\Kq (Nass)	1.322/1.251 129	1.316/1.251 129	1.260/1.241 35	1.230/1.226 152
calc meas Kq_{max}\Kq (Nass)	1.251/1.322 129	1.251/1.316 129	1.241/1.260 35	1.226/1.230 152
Max. (+) difference N of assembliy (type)	7.23 20 (Г)	8.25 41 (Г)	5.97 120 (Г)	11.01 44 (Г)
Max. (-) difference N of assembly (type)	6.58 4 (Г)	6.52 4 (Г)	6.81 56 (B)	8.08 93 (B)
All assemblies root mean square	3.68	3.48	2.60	4.37
Assemblies with SPD root mean square	2.80	2.08	1.98	4.02
Assemblies without SPD root mean square	3.49	3.26	2.30	4.16

Main results in BIPR-7 calculation, compared with the measured assemblywise power distributions Kq(n) for different moments [FPD] of the Cycle 2 Unit 6

$$\text{Difference} = \frac{Kq_{(n)}^{calc} - Kq_{(n)}^{meas}}{Kq_{(n)}^{meas}} \cdot 100\,[\%]$$

n - location number in 360° sector

ИМИТАЦИОННЫЕ МЕТОДЫ ИССЛЕДОВАНИЯ В РАДИАЦИОННОМ МЕТАЛЛОВЕДЕНИИ

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Проблемы, характеризуЮщие предмет радиационного металловедения и его актуальност, связанны с изменением структуры и свойств применяемых в ядерной и водородной энергетике конструкционных материалов, которых терпят воздействие ниско- и высоко-энергетических корпускулярных и йонных потоков. Поведение этих материалов в процессе эксплуатации определяет оптимальнуЮ конструкциЮ и прочность энергетических установок, в которых они используЮтся. Из этого вытекает и одна из самых важнейших задач радиационного металловедения- изучение изменения свойств после облучения энергетическими потоками металлов и сплавов и вместе с этим умелое применение в этой области известных и создание новых конструкционных металлических сплавов и материалов.

В современном этапе развития общества наука выделяет приоритет энергийных проблем и исследования в области радиационных повреждений в металлах и сплавах ведутся широким фронтом. Часть из них посвященны нарушениям в структуре материала при помощи воздействия заряженных частиц. "Имитационные" методы позволяЮт ускорить процесс получения информации о радиационном дефектообразовании, поскольку в реальных условиях эксперименты длились бы чрезвычайно долго по мере накапливания нужной дозы. Обобщение и анализ собранных до сих пор с исспользованием этих методов данных вовсе не тривиальнуЮ задачу. Затруднительно нахождение кореляции с реальными реакторными условиями по мере часто неопределеного исходного структурного состояния пробных тел и реальных конструкции. В этом направлении нужны значительно большие научные усилия.

Источниками йонизируЮщих излучений и йонизованных корпускулярных потоков являЮтся атомные и термоядерные реакторы, радиоактивные изотопы, ускорители йонов, электрические разряды в вакууме, электронные микроскопы, и т.д. Все они предоставляЮт возможность при помощи удачно подобраных и разработанных методик для инициирования "имитационным путем" радиационных повреждений в структуре металлов и сплавов. Эти дефекты в кристалической решетке идентичны с дефектами неутронного облучения и тоже вызываЮт изменение физических свойств материалов: электросопротивления, термо э.д.с., параметра решетки, теплофизических характеристик, диффузионной подвижности атомов. Определение характерных параметров этих свойств дает информациЮ о степени дефектности изучаемых структур. А все изменения механических свойств конструкционных материалов в результате этих процессов информируЮт об их эксплуатационных характеристик в условиях непрерывной работы.

Проникновение и накопление водорода в корпусах реакторов тоже является проблемой, связанной с эксплуатационной надежностьЮ, и изучается специалистам в области радиационного металловедения. Источники водорода и его изотопов в термоядерных реакторах является плазменая рабочая среда. В установках типа ВВЭР существуЮт следуЮщие источники водорода: радиолитическое разложение охлаждаЮщей воды, амиака и гидразин-гидрата; коррозия материалов первого контура; диссоциация водорода в металле в молекулярном виде или в виде метана; ядерные (n,p) реакции. Водород таким образом влияет на механические свойства реакторных материалов, вызывая водородное охрупчивание. Известны разные экспериментальные методы для реализации проникновения водорода в металлах и сплавах и для исследовании его последствия. Результаты применения таких методов и их интерпретация после уточнения оптимальной кореляции с реальными реакторными условиями предоставляют информацию о состоянии водородной проблемой конкретных материалов, применяемых в энергетике.

В Институте металловедения - БАН велись исследования в области радиационного металловедения с использованием "имитационных методов" для получения структурных дефектов в образцах из новых марок сталей при помощи йонного имплантатора, при помощи тлеющего разряда в водородной и гелиевой атмосферах и при помощи электролитного насыщения водородом. Осуществляли поверхностную обработку пробных тел из конструкционных сталей йонизованым водородом и наблюдали дефектную структуру и микропоры после отжига. Электронный микроскоп регистрировал накопление дефектов в структуре сталей Х25Т, Х18АГ12 и , для сравнения, в Х18Н9Т после воздействия ускоренными (до 70 кэв) гелиевыми йонами. Для азотосодержащих сталей показали, что радиационно-стимулированное нитридообразование для температур до 400°С подавляет появление поры гелия. Исследовали мы и поверхностную йонно-стимулированную эррозию в условиях аномального тлеющего разряда в газовой среде из водорода, гелия и аргона. Результаты показывают степень устойчивости исследуемых материалов в условиях ниско-энергетического йонного воздействия, которые существуют в плазменных энергийных устройствах. По мере изучения водородной проблемой разработали на базе масспектрометра МИ 1201В аппаратуру и методику для количественного определения водорода в металлических сплавах и методику исследования водородопроницаемости и коэффициентов диффузии водорода в конструкционных материалах. Исследовано и влияние йонизации водородной молекулы в тлеющем разряде на коэффициентов диффузии и проницаемости пробных тел. Провели и эксперименты для изучения кореляции между концентрациями водорода, йонизованого электролитным путем, механическими характеристиками и некоторыми физическими свойствами исследованых конструкционных сталей. Разработана методика для определения поверхностной микротвердости до и после йонного воздействия и после проникновения водорода во внешные слои материала. Результаты всех этих исследовании позволяют нам прежде всего оценить возможность разширения области приложимости и технологического применения, конечно после необходимых стандартных испытаний, новых, созданных в ИМ - БАН, конструкционных металлических материалов.

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ВЫСОКОАЗОТИСТАЯ СТАЛЬ С ПОВИШЕННОЙ НАДЕЖНОСТИ ДЛЯ ЛОПАТОК НА ЦИЛИНДРОВ НИСКОГО ДАВЛЕНИЯ ПАРНЫХ ТУРБИН

Чавдар Андреев, П. Ленгарски - ИМ-БАН

При создании энергетических установок большой мощности для тепловых и атомных электростанций существено повысились требования к обеспечению надежности и долговечности работы оборудвания в течение сроков службы не мене 30 лет. В связи с этим возникланеобходимость в разработке высокопрочных сталей для высоконагружеженных деталей энергетических установок. К числу таких деталей относятся рабочие лопатки цилиндров низкого давления (ЦНД) паровых турбин реакторов типа ВВЭР.

Анализ условий работый лопаток ЦНД паровых турбин и их повреждений показали, что надежность работы лопаточного апарата определяется уровнем прочности, пластичности ударной вязкосьти, сопротивление усталости и корозионной стойкости в среде влажного пара. Для лопаток последных ступений ЦНД паровых турбин требуются крупно габаритные лопатки с длиной работчей части более 400 mm. В связи с этим для таких лопаток требуются штампованные заготовки из материала с высоким технологическими свойствами при выплавке, штамповке, итермообработке.

В связи с этим возникла необходимость в разработке стали, обладающей при температуре 20 ^{0}C пределом текучести $R_{02} \geq 900$ MPa относительным удлинение $A \geq 15\%$, относительным сужением $Z \geq 50\%$, в сочетании с ударной вязкости KCU более 588 kJ/m^2 и пределом усталости (σ^{-1}) более 450 MPa. Исследования по свойствам стали проведены совместно сотрудниками НПО ЦНИИТМАШ Москва.

Для основа исследования была выбрана коррозионностойкой высокопрочной стали марки 13Х15Н5АМ3 аустенитно-мартензитного классас содержанием азота до 0,1%. Эта сталь удовлетворяет требованиям,предьявленым к лопаточным материалом для НД паровым турбин, однако она обладает недостаточной структурной стабилности и технологичностью, что затрудняет использование для лопаток с рабочей длиной 500 mm и более.

Выплавка опитных плавок, для разработки марок, стали проводились в ИМ БАН методом литья с противодавления на лабораторной установки типа АИ 10/50, содержащая 10 kg индукционная печь. Промышленные плавки были выплавленные в печи с противодавлением емкостью 0,5 t под давлением 1,6 MPa, с последующим электрошлаковым переплавом в установке ЭШПД - 2.

Химический состав исследованных плавок приведен в табл.1. Содержание углерода варировалось от 0,03 до 0,07%, а содержание азота от 0,17 до 0,22 %. Из каждой лабораторной плавки были изготовлены кованые заготовки образцов сечением 15x15 mm и ф 20 mm, длиной 500 mm.

Для выбора температуры нагрева под закалку были определены критические точки при прямом и обратном превращении $\alpha \rightarrow \gamma$ по кривым изменения твердости после закалки и отпуска при различных температурах, приведенным в табл.2. Кривые зависимости изменения твердости от температуры закалки в интервале от 650 до 1150 ^{0}C приведени на рис. 1. Кривые зависимости изменения твердости от температуры отпуска в интервале 350 до 750 ^{0}C приведены на рис. 2. Анализ полученных результатов показал, что введение азота в исследованных пределах оказывает влияние на фазовые превращения в стали. При этом наблюдается повышение критических точек A_{C1} и A_{C3} без изменения температуры начала обратного превращения A_H. Все исследованые плавки при закалке в масло упрочнялись за счет образования структуры мартенсита с различным количеством остаточного аустенита, в зависимости от степени легирования углеродом и азотом.

При отпуске в интервале 350 - 450 °C во всех плавках наблюдается повишения твердости. При дальнейшем повышении температуры отпуска наблюдается снижение твердости до начала обратного $\alpha \rightarrow \gamma$ превращения. В районе температур обратного превращения наблюдается вновь повышение твердости, которое происходит за счет образования дополнительного мартенсита и упрочняющих частиц из частично распадающегося аустенита, образовавщегося во время нагрева при отпуске. По кривым изменения твердости при отпуске были выбраны температуры отпуска, обеспечивающих твердости 340-360 HB.

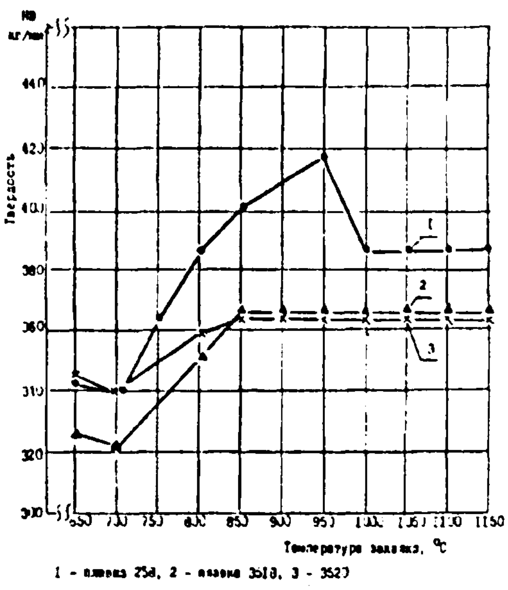

Рис. 1 Зависимость твердости от температуры закалки

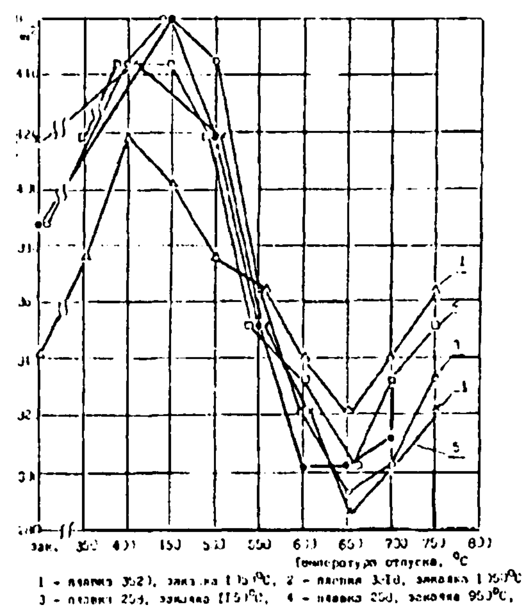

Рис. 2 Зависимость твердости от температуры отпуска

Механические свойства плавок были изучены после закалки при температуры 950, 1050 и 1150 °C и отпуска при температурах от 400 до 750 °C. Механические свойства металла приведены в табл.3. Механические свойства металла промишленных плавок 258 и 310 исследованы на заготовки лопаток полученые после ковки нарадияльноковачной машине в НПО "Бл. Попова". Штамповку заготовок лопаток осуществен на электровинтовом прессе в ПО ЛЗТЛ (гр. Ленинград). Свойства металла приведени в табл. 4. Полученные результати исследований позволяют установить следующее. Наиболее благоприятное сочетание характеристик прочности, пластичности и ударной вязкости стали 05Х15Н4АМ2 получены после закалки 1050 °C и отпуска при температуре 550 °C.

ВЫВОДИ :

1. Разработана новая марка стали типа Х15Н4М3 со содержанием азота в количестве 0,11 - 0,20 %, содержание углерода $\leq$ 0,05% и ограничении суммы углерода и азота в пределах 0,16 - 0,26% [5]. Этот состав позволяет получить металл с аустенитно-мартенситной структурой с количеством аустенита 10-30% .

2. После термообработки эти стали имеют при 20 °C предел текучести $R_{0.2} \geq$ 850 MPa, относительное удлинение A $\geq$ 15%, относительное сужение Z $\geq$ 50% и ударная вязкость KCV $\geq$ 588 kJ/m^2 при критической температуре хрупкости КТХ не выше минус 40 °C. Это в сочетании с высокой коррозионно-механической

прочностью и сопротивлением усталости позволяет использовать их в качестве материала для рабочих лопаток ЦНД паровых турбин, работающих в агресивных средах в замен мартенситной стали 15Х11ФМ.

3. Проведено промышленное опробвание с применением выплавки по технология литья с противодавлением и ЭШПД, разработаной в Болгарии, новой высокопрочной стали марки 05Х15Н5А2М2, из которой изготовлены прутки ф 100 mm и заготовки штампованных лопаток ЦНД длиной 500 mm.

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Таблица 1

Химический состав исследованных лабораторных и промишленных плавок стали типа Х15Н5М3

Номер плавки	Маса плавки kg	Содержание елементов, %							Сумма
		C	Si	Mn	Cr	Ni	Mo	N	C+N
Заданный состав 05Х15Н5А		≤ 0,05	≤ 0,70	≤ 0,80	13,5 15,5	4,0 5,0	1,50 3,00	0,11 0,22	0,16 0,26
3518	10	0,03	0,60	0,81	15,4	4,90	2,93	0,17	0,20
3520	10	0,03	0,53	0,71	14,8	4,98	2,78	0,19	0,22
310	500	0,06	0,20	0,47	14,2	4,26	1,92	0,17	0,23
258	500	0,04	0,19	0,34	13,8	4,80	2,92	0,22	0,26

Таблица 2

Критические точки исследованных плавок стали типа 15Н5М3

Номер плавки	Содержание в %		Критические точки, °C		
	C	N	A_{C1}	A_{C3}	A_K
3518	0,03	0,60	700	850	660
3520	0,03	0,53	700	850	650
310	0,06	0,20	700	850	650
258	0,04	0,19	710	950	650

Таблица 3

Механические свойства металла лабораторных и промышленных плавок стали типа Х15Н5М3

номер плавки	Режим термообработки	R_{02} MPa	$R_ъ$ MPa	A %	Z %	KCU kJ/m^2
Технические требования к лопаткам ЦНД		800	900	15	50	588
3520	Закалка 1050 ^{0}C, 1h отпуск 400 ^{0}C, 4h	1129	1500	23.5	56,7	430
258	Закалка 1050 ^{0}C, 1h отпуск 550 ^{0}C, 4h	1100	1297	16,7	60,0	1170
310	Закалка 1050 ^{0}C, 1h отпуск 550 ^{0}C, 4h	950	1125	17.0	58,9	1160

Таблица 4

Механические свойства металла заготовок лопаток. Термобработка закалка 1050 ^{0}C, 1h, масло; отпуск 550 ^{0}C,4h,воздух

Номер плавки	Место вырезки образцов	Режим термо-обработки	R_{02} MPa	$R_ъ$ MPa	A %	Z %	KCU kJ/m^2
258	замок перо	закалка 1050 ^{0}C,1h масло	967 984	1137 1174	15,2 15,8	48.6 54,5	1100 1120
310	замок перо	отпуск 550 ^{0}C,4h воздух	945 965	1131 1093	15.0 15,8	49,0 57,8	1284 1343

BG9600420

ВЫСОКОАЗОТИСТАЯ СТАЛЬ ТИПА 38 ХНЗМАФА ДЛЯ КРЕПЕЖНЫХ ДЕТАЙЛЕЙ РЕАКТОРОВ АЭС

РАШЕВА И. А., АРГИРОВ Х. А., СТОЙЧЕВ Т. В. - ИНСТИТУТ МЕТТАЛОВЕДЕНИЯ БОЛГАРСКОЙ АКАДЕМИИ НАУК

При создании энергетических установок большой мощности для атомных электростанции существенно повысилис требования к обеспеченийо надежности и долговечности работы оборудования. В связи с этим возникла необходимость в разработке перлитной стали с повышенной прочностью и сопротивлением хрупкаму разрушенийо для высоконагруженных изделий энергооборудования, к числу которых отнасятся крепежные детали.

В результате научно-технического сотрудничества между НПО ЦНИИТМАШ - Москва и Институтом металоведения Болгарской академии наук была создана новая высокопрочная сталь типа 38 ХНЗМАФА перлитного класса с повышенным содержением азота [1].

Азот является важным легирующим элементом, который оказывает существенное влияние на структуру и свойства перлитных хромомолибденованадиевых сталей и может быть использован для карбидно-нитродного упрочнения этих сталей взамен карбидного упрочнения.

Возможность легирования стали азотом в различных концентрациях обусловена методом выплавки. В ИМ-БАН разработан метод литья с газовым противодавлением в атмосфере азота, позволяющий выплавлять стали со сверхравновесным содержанием азота. Этим методом выплавляются промишленые слитки массой 400 и 2 000 кг, которые перековываются на заготовки нужного размера.

Сталь производится и на установках электрошлакового переплава под давлением, создающих возможность легирования высокой концентрацией азота и получение слитков с высокой чистоты и низкой ликвацией.

Основные характеристики свойств и технологии термической обработки представлены в таблицах 1-6 и на рис. 1-6.

Таблица 1
Химический состав, вес %

Сталь	C	Si	Mn	Cr	Ni	Mo	V	W	N
34NiCrMoV145N1	0.23 0.32	0.17 0.37	0.25 0.50	1.20 1.50	3.0 3.5	0.35 0.45	0.10 0.18	-	0.05 0.07
34NiCrMoV145N2	0.10 0.30	0.17 0.37	0.25 0.50	1.50 3.0	2.0 3.0	0.30 0.50	0.10 0.20	0.05 0.15	0.15 0.20

Безазотные аналогии: сталь 34NiCrMoV145 (DIN); сталь 38ХН3ПФА (ГОСТ); сталь 4337 + V (AISI).

Таблица 2
Механические свойства при 20 ^{0}C после улучшения (зак. 900 ^{0}C, отпуск 580 ^{0}C)

Сталь	$R_{0.2}$ MPa	R_m MPa	A_5 %	Z %	KCU MJ/m^2	HCR
34NiCrMoV145	880	980	14.0	40.0	0.588	32
34NiCrMoV145N1	1045	1134	15.0	62.6	1.03	31
34NiCrMoV145N2	1060	1160	16.0	68.0	1.20	37

Таблица 3

Предел текучести при 350 °C, критическая температура хрупкости и параметр статистической трещиноустойчивости

Сталь	34NiCrMoV145N1	34NiCrMoV145N2	34NiCrMoV145
$R_{0,2}$, MPa	970	930	735
KTk, °C	-10	-70	-80
KIC, MPa	110	108	102

Таблица 4

Физические свойства и критические точки

Сталь	Гидростатическая плотность, g/cm^3	Теплопроводность при 400 °C, W/m°C	$A_{c.1}$ °C	$A_{c.3}$ °C
34NiCrMoV145N2	7.850	24.80	695	800
34NiCrMoV145N1	7.850	24.80	700	800
34NiCrMoV145	7.850	24.50	700	760

Таблица 5

Режимы предварительной термической обработки

Виды термической обработки	Температура нагрева, °C	Температура изотермической выдержки, °C	Скорость охлаждения, °C/h
Отжиг с непрерывным охлаждением	800-840	-	не больше 30
Отжиг с изотермической выдержкой	800-840		не больше 30
Высокотемпературный отпуск	680-700	-	с печью

Таблица 6

Режимы окончательной термической обработки

Закалка			Отпуск		
°C	среда	HRC	°C	среда	HRC
800-850	масло	45-46	600	воздух	32-35

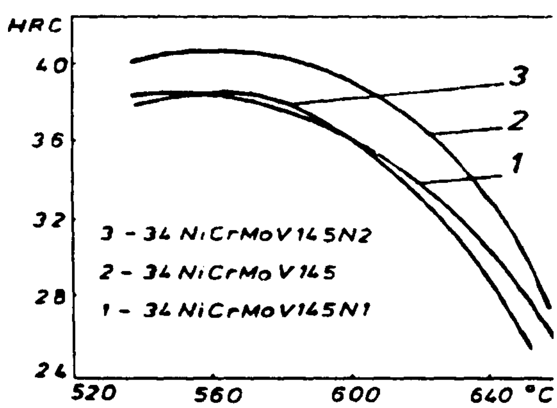

Рис. 1. Твердость при различных температурах отпуска

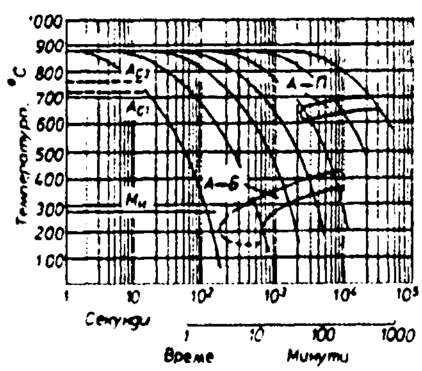

Рис. 2. Анизотермическая диаграма распада переохлажден-ного аустенита стали 34NiCrMoV145N1

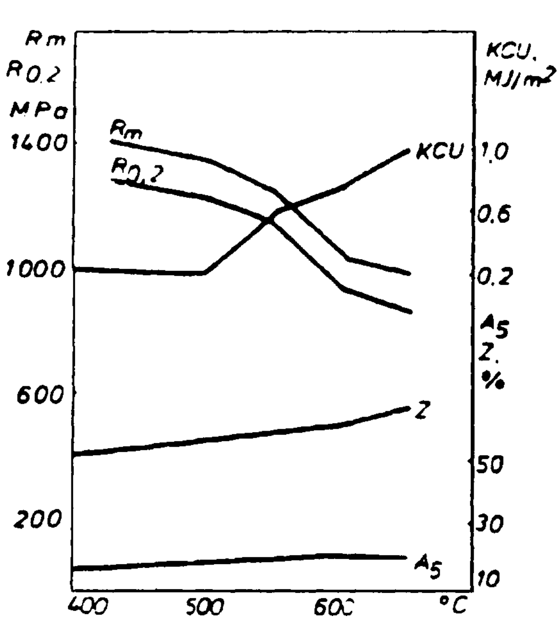

Рис. 3. Механические свойства при различных температурах отпуска (зак. 900°C, масло)

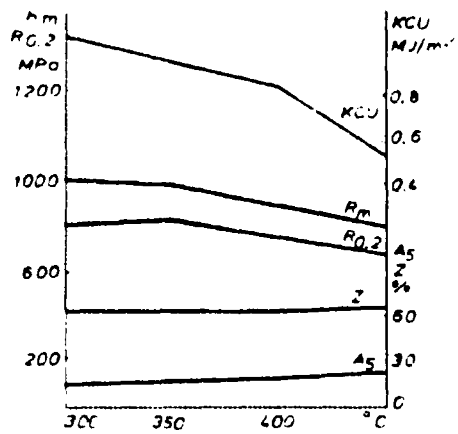

Рис. 4. Механические свойства при различных температурах испытания (зак. 900°C, масло, отп. 600°C)

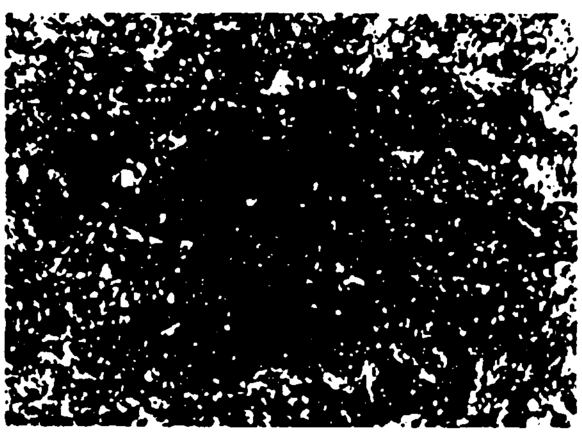

Рис. 5. Микроструктура стали 34NiCrMoV145N1 в доставном состоянии . x250

Рис. 6. Микроструктура стали 34NiCrMoV145N1 в улучшенном состоянии . x500

Температурный интервал ковки 1050-850 °C, охлаждение после ковки - замедленное в изолированных камерах.

Параметры обрабатываемости высокоазотных сталей того же порядка как и при безазотных аналогох.

Стали могут быть изпользованы в качестве релаксационностойкого материала в ряде областей техники: энергетическое машиностроение (роторы, турбин, валы компрессоров), ответственные детали машин, высоконагруженные детали гидравлических прессов, элементы крепления реакторов в АЭС, конструкционный материал для химической промышленности, инструменты (штампы, оправки) для работы в температурном интервале от +350 до -90 °C.

Сталь производится в виде сортового кованного проката следующих размеров:

- круг: диаметр 90-300 mm, длина до 9000 mm;
- квадрат: от 20x20 до 90x90 mm;
- прямоугольник: (50-120) x (20-50) mm;
(100-150) x (60-100) mm;
(150-200) x (100-200) mm;
(200-250) x (200-250) mm;
(250-300) x (250-300) mm;

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EXPERIMENTS FOR NEUTRON FLUENCE ASSESSMENT ON VVER-440 AND VVER-1000 PRESSURE VESSEL

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These researches have been carried out under the Kozloduy NPP contracts.

ABSTRACT: *The activity of shavings sampled out from the inner pressure vessel wall of Unit 1 after the 14th cycle and detectors displaced in channel INEI of Unit 3 of Kozloduy NPP have been measured. The measured shavings activities from weld 4 of Unit 1 provided the opportunity to compare the experimental results with the calculated ones at the expected maximum embrittlement position. The calculations of the expected activities have been carried out taking into account the local power distribution. Comparisons of calculated and measured activity values have indicated that the computed values are lower about 20%.*

I. COMPARISON OF MEASURED AND CALCULATED ACTIVITIES OF SHAVINGS

Determination of the activities of the shavings from the Reactor Pressure Vessel (RPV) of Kozloduy NPP Unit 1 has been carried out.

The shavings have been sampled out from weld 4 at 0^0 relative to 60^0 sector of symmetry, after the 14-th operated cycle in 1989. Unit 1 had been working with standard core loading and fresh fuel assemblies on the core periphery for the first 10 cycles; with burn-up fuel on the periphery of the core during cycles 11 through 12, and with 36 dummy cassettes in the periphery during cycle 13 and 14.

The analysis of gamma spectra of 54 Mn in the shavings, resulting from the 54 Fe(n,p)54 Mn reaction, was carried out using a low background spectrometer with 1.9 keV energy resolution of 60Co. The shaving materials were properly diluted, and aliquotes of the solution were measured in 450 ml MARINELLI beakers. The measured activities of the shavigs taken from different sites of the RPV weld referred to 01.01.1993 are presented in Table 1.

The neutron fluxes in accordance with real operation have been calculated by three dimentional diffusion code PYTHIA[1] and interface code REAC440[2], by three dimentional synthesis method using DORT[3] discrete ordinates code, FLUNG[4] neutron constant library and IRDF-90[5] activation data for 54Fe. The activity estimation has been done by ACTIVAT code[6].

Table 1. The measured and calculated activities of the shavings referred 01.01.1993.

Place No	Depth mm	Activity, E+5 Bq/g Measured	Calculated
(2)-0^0	3.4-4.1	1.34 ± 0.04	1.14
(2)-0^0	4.1-4.8	1.32 ± 0.04	0.984
(3)-0^0	4.0-4.8	1.25 ± 0.04	0.984

The comparison of measured and calculated results of shavings activities shows a discrepancy about 20 %. However the more correct calculation of the activity taking into account the local power distribution during the operation must diminish the calculated activity values. That is why more studies have to be done to validate the calculational results.

II. EXPERIMENTS IN INEI CHANNEL

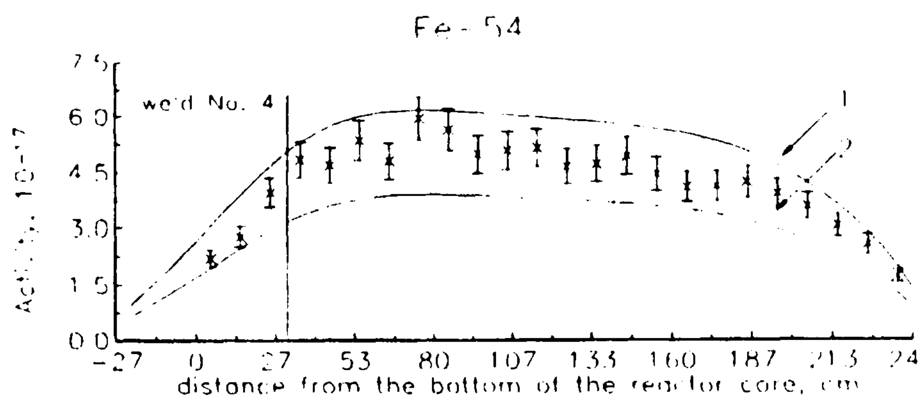
Preliminary experimental measurements have been carried out by means of string detectors of Fe and Cu along the INEI channel and foil detectors also made of Fe and Cu, alocated at the level of weld 4 in a channel at 0° relative to 60° sector of symetry of Unit 3, NPP-Kozloduy. The detectors have been irradiated twice in 10 days. The relative core height distribution, determined by the gamma lines of 54Fe, 58Fe, 63Cu is shown in Fig.1. The reaction rates of 54Fe and 63Cu are presented in Table 2.

The neutron fluxes in accordance with real operation have been calculated by three dimentional diffusion code SPPS[7], by two dimentional DOT4.2 [8]discrete ordinates code, FLUNG neutron constant library and IRDF-90 activation data for 54Fe, 63Cu. The neutron source distribution for 56 % operated power have been prepared by REAC440 code. The activity estimation for the time of irradiation has been done by ACTIVAT code.

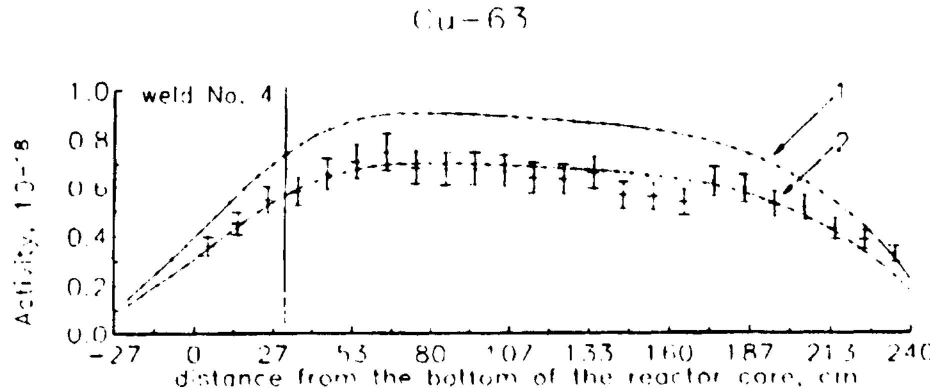
Table. 2. Reaction rates [s^{-1}] of the detectors irradiated in the channel of INEI, Unit 3 of Kozloduy NPP

Reaction	Experiment		Calculation	
	I	II	Min	Max
$^{58}Fe(th)$	(6.17±0.80)E-15	(5.97±0.34)E-15	-	-
$^{54}Fe(n,p)$	(3.60±0.18)E-17	(3.52±0.17)E-17	3.40E-17	2.45E-17
$^{63}Cu(n,\alpha)$	(6.10±0.31)E-19	(5.38±0.27)E-19	5.00E-19	3.88E-19

The distributions of the reaction rates along the height calculated relative to the innermost (min) and outermost (max) point of INEI in radial direction are presented in Fig.



Axial neutron flux distribution, block 3, cycle 11, 11.08.92 - 21.08.92

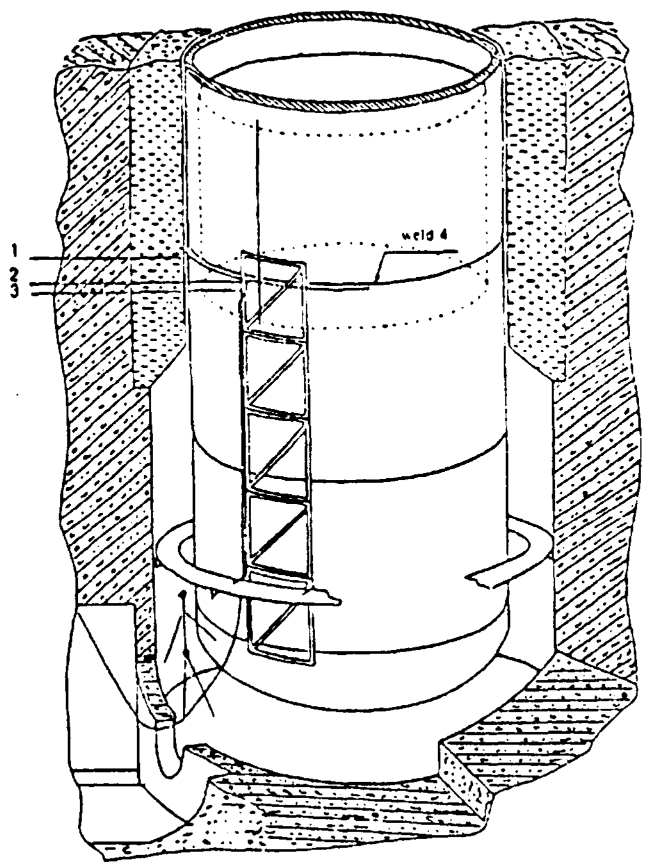


Axial neutron flux distribution, block 3, cycle 11, 11.08.92 - 21.08.92

Fig.

III. EX-VESSEL EXPERIMENTS

To verify the calculation results ex-vessel measurements of neutron fluence with threshold activation detectors will be carried out in the air gap between the vessel and the thermal insulation. A special device to carry the set of enriched foil of isotope detector 63Cu and chemical pure Fe and Nb foils, has been placed behind the vessel wall. It is composed of two aluminium pipes containing iron and copper wire detectors. The pipes are situated perpendicularly to each other so that one of them covers a 60^{o} sector in azimuthal direction of the reactor core. Such measurements of the fast neutron fluence will be carried out on Units 1, 2, 3, and 4 of Kozloduy NPP.



Ex-vessel rack device with activation detectors for VVER-440/230

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BG9600422

ABOUT RELIABILITY ON VVER PRESSURE VESSEL NEUTRON FLUENCE CALCULATION

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ABSTRACT. *Reliability study of neutron fluence calculation for VVER pressure vessel has been carried out. The estimation of influence of the geometry approximation in calculation model and the choice of neutron cross section data file on the calculation results are presented.*

The neutron fluence determination on the reactor pressure vessel is an essential part of the Surveillance Program for ensuring save operation of the reactor unit during the lifetime limit.

The main tools for determining the neutron flux are the neutron transport calculations. The reliable assessment of the neutron flux and its responses in the vicinity of the reactor pressure vessel is a difficult problem for the reactor systems. It is caused by the complexity (multi-layer heterogeneous structures) of study media and its considerable extension (15-20 free mean paths). The adequate modelling of the neutron transport is restricted by the proximate description of the complicated interaction of the neutron with the nucleus (the cross section data libraries must be widely tested for such calculations). and limited abilities of the existing transport codes and the used computers..

In this paper the estimation of influence of: 1) the geometry approximation in the calculation model; and 2) the choice of the neutron cross section data files; on the calculation results are presented.

1. TORT APPLICATION IN RPV NEUTRON FLUX CALCULATIONS

The neutron flux values onto the Pressure Vessel(PV) of VVER-1000 and VVER-440 reactors, at the places (Table 1) important for the metal embrittlement surveillance, have been calculated by 3D code TORT [1] and synthesis method [2] by code DORT [3], both based on the discrete ordinate Sn method. The comparison of the results received by both methods has been applied for a demonstration of the reliability of synthesis method.

The neutron fission source is represented by the power efficiency, averaged over the plane cross section of each cassette. The axial dependence is taken into consideration. The library FLUNG [4] has been applied in all following flux calculations.

The comparison of the VVER-1000 flux values (Table 2) shows a good consistency within the limits of the solution accuracy. The differences diminish from 5% to 1% with enlarging the energy range limits.

These results are expected for VVER-1000, because the places of interest are far from the reactor core axial edges.

TABLE 1. Places for the Test Comparison

Reactor	N°	r, cm	θ,°	z, cm	Comments
VVER-1000	1	207.35	8	*96.0	on RPV, azim max, axial max
	2	207.35	30	96.0	on RPV, azim min, axial max
VVER-440	1	178.40	30	29.5	on RPV, azim max, weld 4
	2	192.45	30	29.5	beh. RPV, azim max, weld 4
	3	178.40	13	29.5	on RPV, azim min, weld 4
	4	192.45	13	29.5	beh. RPV, azim min, weld 4

* z=0 - core bottom

The more important conclusion is connected with the good consistency of the flux values for VVER-440 in the places near the core bottom, the weld seam 4 (Table 3).

In addition, it may be noted that the CPU time for the synthesis method calculations is about 18 times shorter than the TORT one. The accomplished comparison indicates that at reasonable cost the calculations necessary for the Metal Embrittlement Surveillance Program should be performed by the synthesis method.

This comparison, however, is not sufficient for estimating the inaccuracy of the neutron flux calculation results because it is based on one and the same calculation method of discrete ordinates.

TABLE 2. Neutron Fluxes Ratio for VVER-1000

Point	Energy range, Mev			
	>5.0	>3.0	>1.0	> 0.5
1	1.03	1.02	0.98	0.99
2	1.05	1.03	1.01	1.01

TABLE 3. Neutron Fluxes Ratio for VVER-440

Point	Energy range, Mev			
	>5.0	>3.0	>1.0	> 0.5
1	0.971	0.969	0.970	0.974
2	0.951	0.957	0.966	0.964
3	0.960	0.961	0.968	0.970
4	0.940	0.950	0.963	0.967

2. IRON SPHERE BENCHMARK

Calculation results in according to the Iron sphere benchmark [5] carried out in Check (SKODA) have been obtained. The ^{252}Cf fission source is placed in the centre of on iron sphere with outer radius of 25 cm. The spectral measurements of the neutron leakage from the sphere have been done using a proton-recoil detector and stilbene crystal spectrometer located at 1 m from the source. There are presented the flux responses above different energy limits. Comparisons of experimental leakage spectrum with the calculated ones obtained by the multigroup neutron cross sections based on ENDF/B-4 and ENDF/B-6 data have been carried out (Fig.1, Table 4).

In the energy region above 1 MeV, of the greatest interest for the neutron embrittlement of reactor vessel, the best consistency with the experiment has been obtained by the data from ENDF/B-6 in VITAMIN-E [5] group structure. The calculations performed by the DLC37F multigroup library (from ENDF/B-4) underestimate the experimental ones. This is in accordance with the another authors results [5,6]. The differences between the experimental and calculated by FENDL [7] (from MAAE) results show that the mentioned version is not good enough for transport calculation.

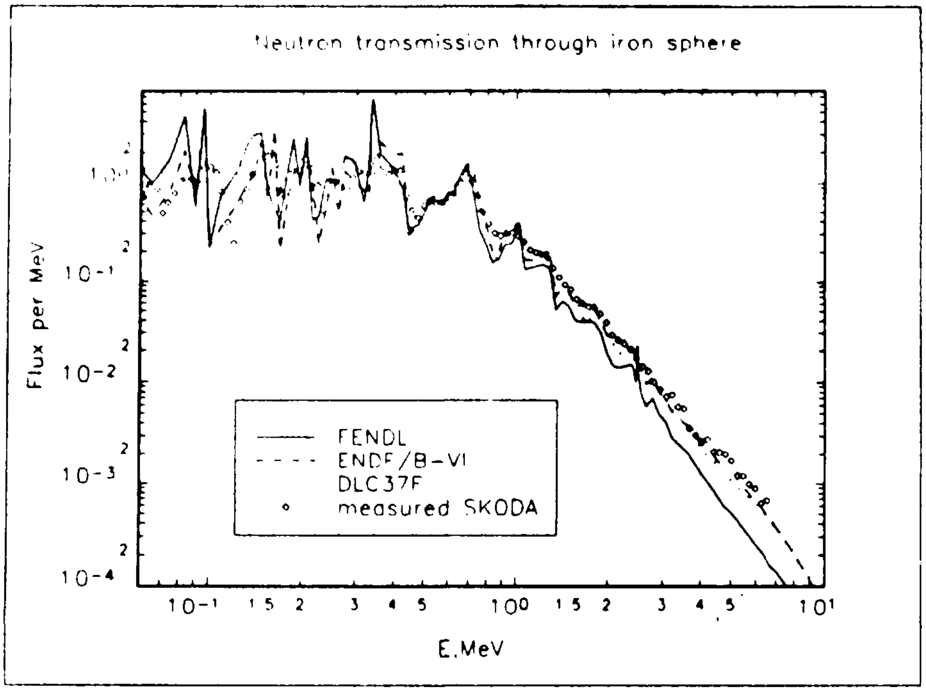

Fig. 1

TABLE 4. Ratio C/E of calculated to experimental SKODA flux responses

Above Energy, MeV	FENDL	DLC37F	ENDF/B-VI
0.1	1.0209	0.9980	1.1107
1.0	0.6412	0.8036	0.9169
2.0	0.5235	0.8152	0.8796
3.0	0.4007	0.7427	0.8021
4.0	0.4639	0.6969	0.7533

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BG9600423

CALCULATIONAL RESULTS FOR RADIATION EMBRITTLEMENT OF VVER PRESSURE VESSEL AT KOZLODUY NPP

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ABSTRACT: Calculation procedure for radiation embrittlement determination of the reactor pressure vessel of WWER type of reactor on the basis of the neutron fluence has been developed. The radiation state of all pressure vessels of the NPP Kozloduy reactors has been determined.

Four reactor of type VVER-440/230 and two reactors VVER-1000/320 are in operation at the Kozloduy NPP. For the reactors VVER-440/230 it has not been constructively foreseen a special place for disposition of sample-witnesses (SW), i.e. the determination of metal radiation state is possible only by application of calculation methods. On the contrary, in the reactors of type VVER-1000 it has been foreseen by project and there are placed in it certain amount of SW. The spots, where the SW are situated, are high above the active core. In order to assess the base metal embrittlement in the most overloaded area and in the weld metal, the results of experimental tests of irradiated in such way SW should be referred to the critical spots of interest by a speciallly developed procedure. This is why, also in this case, the appliance of a calculation procedure, accounting for the change in critical temperature of neutron brittleness by the neutron fluence, is compulsory.

Calculational procedure for evaluation of the shift of critical temperature of embrittlement on the basis of neutron fluence has been applied for assessing the reactor pressure vessel (RPV) embrittlement and the prognosis of RPV life time for VVER-440/230 and VVER-1000/320. The calculated results are lower than the project values, because the real fuel regimes, the low leakage schemes and loadings with dummy cassettes have been taken in consideration in neutron fluence calculation. The temperature of the outer wall of RPV of the VVER-440/230 have been measured. No significant deviation have been observed. This guarantees the correct application of the calculational procedure.

Table 1. The ratio Tk^{real}/Tk^{proj} for the VVERs-440/230 RPV, Kozloduy NPP

Unit	30° before annealing	13° after annealing
Unit 1	1-14 cycle ---------- 115/129=0.891	15-16 cycle ---------- 46/52=0.884
Unit 2	1-16 cycle ---------- 162/175=0.925	17 cycle ---------- 72/78=0.923
Unit 3	1-8 cycle ---------- 122.4/140.5=0.871	9-10 cycle ---------- 47.36/55.3=0.856
Unit 4	1-10 cycle ---------- 63.2/69=0.915	

Table 2. Neutron fluence and ΔT_F on the base metal for the maximum overloaded direction, Unit 5, Kozloduy NPP

cycle No	3D calc. F, $10^{18}cm^{-2}$	3D calc ΔT_F, °C	project F, $10^{18}cm^{-2}$	project ΔT_F, °C
1	1.140	12.5	1.425	13.5
2	2.346	15.9	1.425	17.0
3	3.452	18.1	1.425	19.5

Table 3. Neutron fluence and ΔT_F for the weld metal, Unit 5, Kozloduy NPP

cycle No	8°C F, $10^{18}cm^{-2}$ W3	8°C F, $10^{18}cm^{-2}$ W4	8°C ΔT_F, °C W3	8°C ΔT_F, °C W4	30°C F, $10^{18}cm^{-2}$ W3	30°C F, $10^{18}cm^{-2}$ W4	30°C ΔT_F, °C W3	30°C ΔT_F, °C W4
1	1.078	0.909	20.5	19.4	0.362	0.298	14.2	13.4.
2	2.218	1.933	26.1	24.9	0.754	0.653	18.2	17.4
3	3.301	2.950	29.8	28.7	1.129	1.006	20.8	20.0

Table 4. Neutron fluence, shift of the ΔT_F and T_K for VVER-1000, Unit 6, Kozloduy NPP

place	8°C F, $10^{18}cm^{-2}$	8°C ΔT_F, °C	8°C Tk, °C
weld No 3	1.079	20.5	-19.5
weld No 4	1.006	20.0	+10.0
base metal	1.110	12.4	-32.6

Table 5. The mean temperature values of outer wall of VVER-440/230, Kozloduy NPP

Unit	Tin, °C		Tw4, °C	
	passp.	measur.	passp.	measur.
Unit 2	270	269±4	250	255±6
Unit 4	270	270±3	250	254±5

Fig.1 Shift of TF, Weld 4, 30 deg, NPP Kozloduy, Unit 1

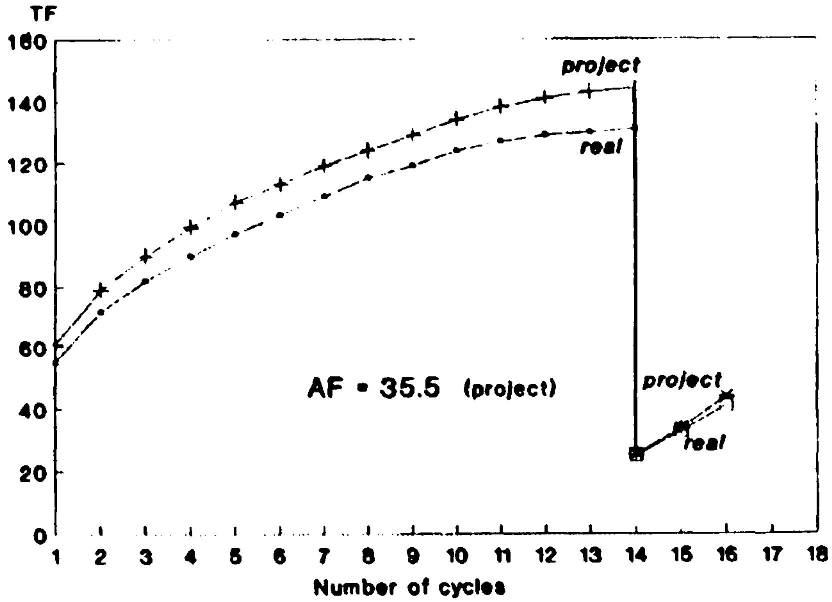


Fig.2 Shift of TF, Weld 4, 30 deg, NPP Kozloduy, Unit 2

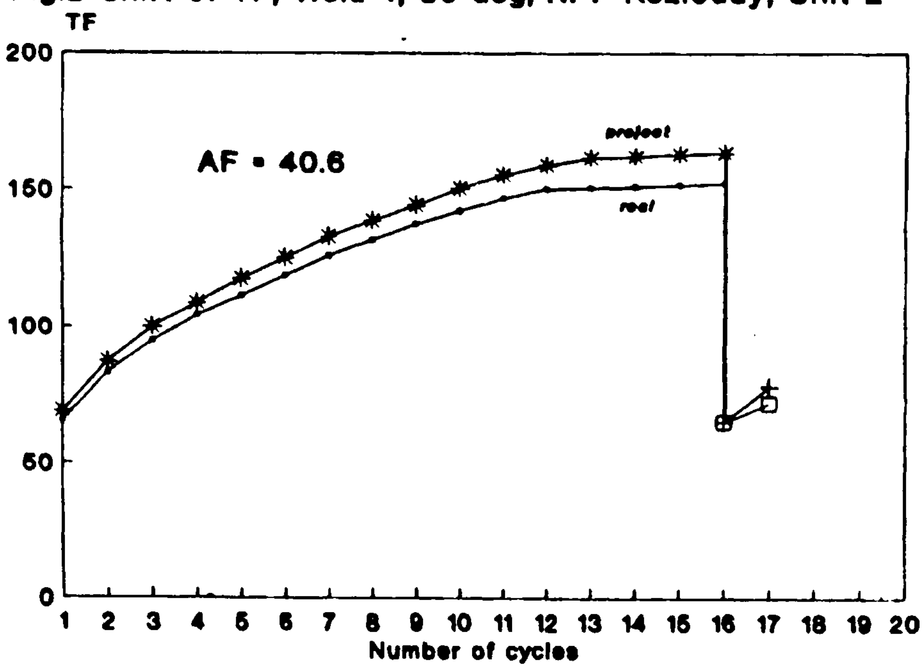


Fig.3 Shift of TF, Weld 4, 30 deg, NPP Kozloduy, Unit 3

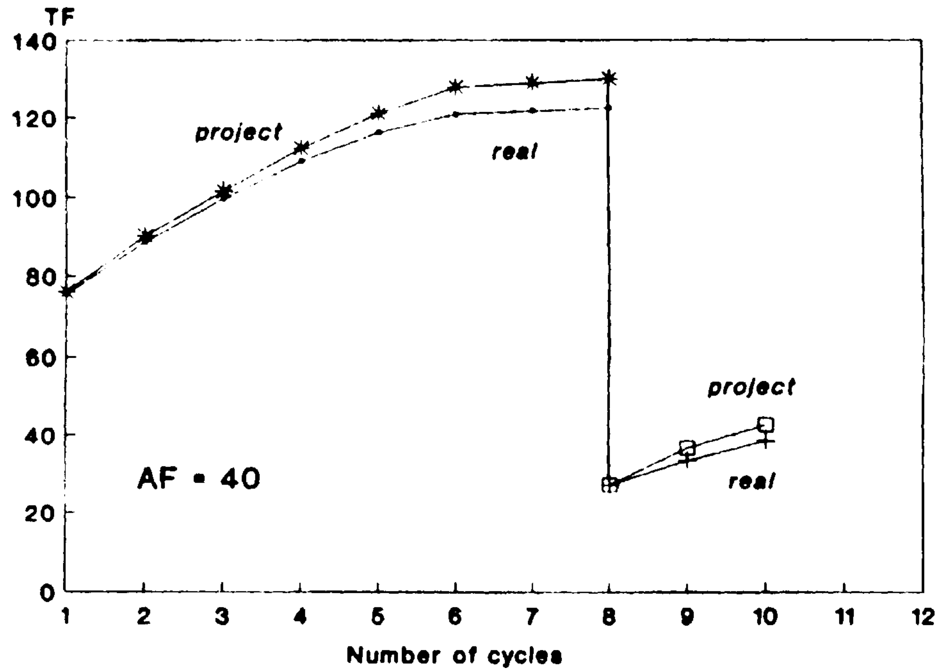
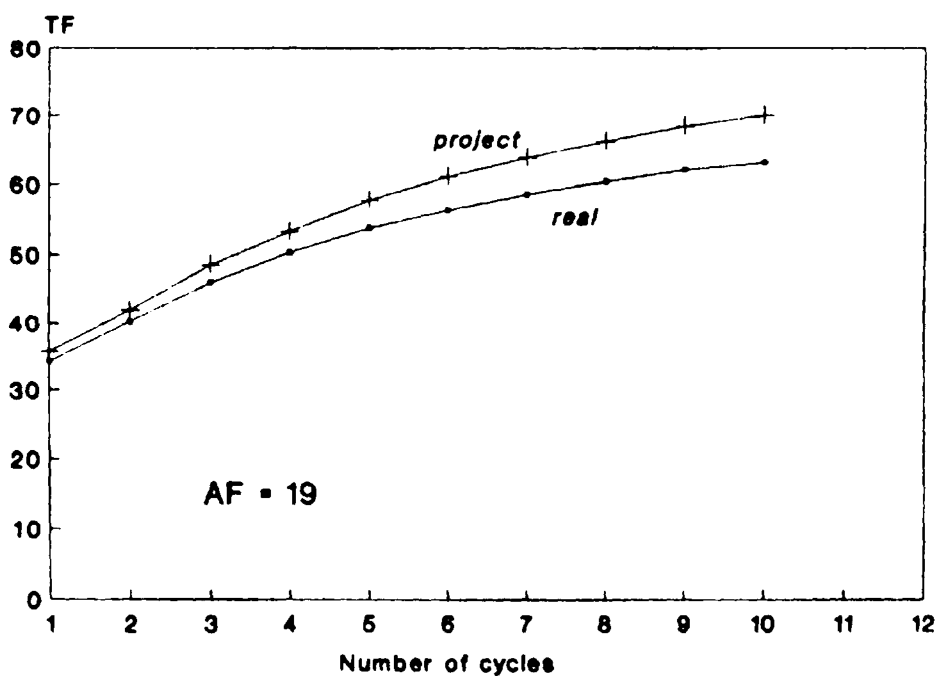


Fig.4 Shift of TF, Weld 4, 30 deg, NPP Kozloduy, Unit 4



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АВАРИЙНОЕ РАСХОЛАЖИВАНИЕ РЕАКТОРА ВВЭР-440 С ПОМОЩЬЮ ПРОЦЕДУРЫ "ПОДПИТКА - ПРОДУВКА"

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Полная потеря питательнои воды и невозможность восстановления системы АПЭН может привести к серьезным последствиям для активной зоны. При полном обезпаривании ПГ в процессе отвода остаточного тепловыделения через ПК ПГ или БРУ-А идет постепенное разогревание I контура. После осушения ПГ, теплоноситель в I контуре достигает насыщенного состояния. В процессе разогревания теплоносителя I контура достигается уставка ПК КД, который начинает циклично открывать и закрывать. Это приведет к потере массы в I контуре с последующим оголением и плавлением ТВС, если отсуствует возможность подпитки I контура системой аварийной подпитки (НАП).

Оператор может стартировать процедуру "Подпитка-продувка" с целью расхолаживания зоны в случае полной потери ПВ. Эта процедура требует включение НАП оперативным персоналсм для восстановления массы теплоносителя в I контуре и принудительного открытия ПК КД, для отвода остаточного тепловыделения в бокс ПГ. Основная идея процедуры Feed and Bleed (F&B)-поддержание массы теплоносителя в I контуре на определенном уровне, низкого давления и температуры в реакторной установке (РУ).

Расхолаживание с помощью F&B-очень ответственное действие оперативного персонала с целью резко уменьшить вероятность плавления активной зоны при полной потере ПВ. Исследования, приведенные в последное время, показывают уменьшения вероятности плавления зоны порядком 1,4. раза для реакторов PWR.

Время для стартирования F&B-самый важный параметр для предотвращения плавления зоны. Согласно оценкам и анализам, проведенных для реакторов типа ВВЭР-440 (В 230) , это время определяется достижением температуры в горячих нитках около 315° С и составляет~3 часов.

I. Анализ аварии с полной потерей питательной воды

В этом разделе приведены результаты анализа двух возможных сценариев развития аварии "Полная потеря ПВ" без вмешательства оперативного персонала:

- срабатывание АЗ по сигналу "обесточивание АЭС";
- срабатывание АЗ по сигналу "низкий уровень в ПГ".

Для проведения указанных расчетов принят модел РУ, использованный в [1]. Исходные условия и параметры блока соответствуют тем, принятым для анализа аварийного события "Потеря питательной воды" [1], исключая:

- более детального моделя ПГ;
- обесточивания АЭС в начале аварии, а не в моменте достижения низкого уровня в ПГ (для первого сценария аварии).

I. 1. Полная потеря ПВ с обесточиванием АЭС в начальном моменте.

При этом сценарии предполагается, что потеря ПВ вызывается обесточиванием АЭС, а аварийная подпитка ПГ не восстанавливается из-за механических повреждений.

Этот анализ проведен исходя из следущих соображений:

- сделать сравнение с подобным расчетом, проведенный с программой ДИНАМИКА, ОКБ "Гидропресс", Россия;
- оценка самого благоприятного развития этой аварийной ситуации и получение самые оптимистичные оценки.

В момента t=0 s налицо следующие события:

- полное обесточивание АЭС;
- закрытие АСК;
- начало выбега ГЦН (2 механического и 4 электромеханического);

- прекращение подпитки ПГ ПВ;
- АЗ - 1 по сигналу "обесточивание" с опозданием 0,9 сек.

В результате разогрева I контура ПК КД открывает первоначально в 55-Ю минуту. В результате постоянного разогрева I контура уровень в КД поднимается и он заполняется однофазным флуидом в 4,9 ч. Теплоноситель в I контуре достигает насыщенного состояния в 5,5 ч. с начала транзиента. ПГ перестаЮт эффективно отнимать остаточного тепловыделения в 4,6 ч. с начала аварии, а это приводит к значительному увеличениЮ градиента нарастания температуры теплоносителя в I контуре.

I. 2. Полная потеря ПВ со срабатыванием АЗ по низкому уровнЮ в ПГ

При этом сценарии предполагается, что полная потеря ПВ и аварийная подпитка ПГ вызывается отказом по общей причине в машинном зале. Обесточивание АЭС наступает в момент генерирования сигнала АЗ-1 низкому уровнЮ в 2 из 6 ПГ (-270 мм) по приборам среднего уровня.

Основным отличием поведения блока в этом случае заклЮчается в значительной потери массы ПГ в начальном периоде аварии, которая ускоряет процессы в целом.

Этот сценарий аварии представляет более консервативного возможного способа развития в подобной аварийной ситуации.

ПК КД открывается первоначально в 11-Ю минуту транзиента.

Теплоноситель в I контуре достигает насыщенного состояния в~4,0 ч. с начала аварии.

ПГ перестаЮт отнимать эффективно генерированнуЮ теплоту в 3-ем часу, а заполнение КД наблЮдается в 3,4 ч. с начала аварии.

Температура теплоносителя в I контуре достигает значение 315° С в 3,5 ч., которое соответствует уровнЮ в ПГ 0,35 м. и представляет 17 % от номинального общего уровня ПГ.

Время самых важных событий приведены в Табл. 1.

Развитие аварийного процесса **Таблица 1**

Событие	Время, сек.
Полная потеря питательной воды	0,0
Срабатывание АСК	45,0
Срабатывание АЗ-1	48,0
Обесточивание блока	48,0
Конец электромеханического выбега ГЦН - 310	228,0
Первое открытие ПК КД	640,0
Заполнение КД	12240,0
Прекращение эффективного отвода тепла через ПГ	11400,0
Достижение состояния насыщения в I контуре	14300,0

II. Анализ переходных процессов в РУ с приложением процедуры "Подпитка - Продувка"

Представленные анализы в разделе I даЮт основание выбрать самый неблагоприятный сценарий развития аварии "Полная потеря ПВ".

Результаты расчетов представлены на рис.1-4. Исходные условия для проведенного анализа такие же, как в I. 2. Критерий стартирования процедуры F&B - температура в горячих нитках 315° С.

Открытие ПК КД в 3,5 ч. из-за аварии вызывает резкого падения давления в I контуре до значения, соответствуЮщее давлениЮ насыщения при температуре 315° С, P_{1k}~10,7 MPa. Впрыскивание холодной воды от НАП вызывает постепенного расхолаживания I контура и соответного понижения давления в I контуре. Температура в горячих нитках остается ниже температуры насыщения, за исключением очень короткого периода в начале стартирования процедуры. Недогрев до температуры насыщения достигается в конце расчета около 40°С.

Ожидаемые стационарные параметры блока при этих условиях приложения процедуры F&B следующие-P_{1k}~10 бара, T_{1k}~120° С. Развитие аварийного процесса "Потеря ПВ" с приложением F&B процедуры дано в Табл. 2.

Развитие аварийного процесса с приложением F&B **Таблица 2**

Событие	Время, сек.
Полная потеря питательной воды	0
Срабатывание АСК	45
Срабатывание АЗ-1	48
Обесточивание блока	48
Конец электромеханического выбега ГЦН-310	228
Первое открытие ПК КД	640
Заполнение КД	12240
Стартирование процедуры F&B при следующих параметрах:	12400
- температура в горячих нитках-315° С	
- общий уровень в ПГ - 0,35 м.	
- открытие одного ПК КД	
- пуск двух НАП	
Параметры РУ в конце расчета:	20000
- температура в горячих нитках - 147° С	
- давление в I контуре-10 бар	

ЛИТЕРАТУРА

1. WANO SIX MONTH PROGRAMME. Item C. Accident analysis, WESI, 1992.

Описание рисунок:

Рис. 1. Изменение температуры теплоносителя:

- 9010000 - $T_{вх}$ в реакторе
- 15010000 - $T_{изх}$ из зоны
- 795020000 - T_S в первом контуре

Рис. 2. Изменение уровня в КД

Рис. 3. Изменение массы флуида в ПГ

- SG1 - в одиночном ПГ
- SG1+1 - в двойном ПГ
- SG1+1+1 - в тройном ПГ

Рис. 4. Изменение давления в парогенераторе SG1

Список сокращений

АПЕН - Аварийные питательные электронасосы;
ПГ - Парогенератор;
ПК - Предохранительный клапан;
БРУ-А - Быстродействующая редукционная установка;
КД - Компенсатор давления;
ТВС - Тепловыделяющая сборка;
НАП - Насосы аварийной подпитки первого контура;
ПВ - Питательная вода;
РУ - Реакторная установка;
АСК - Аварийные стопорные клапаны;
АЗ 1 - Аварийная защита I рода;
F&B - Подпитка - продувка.

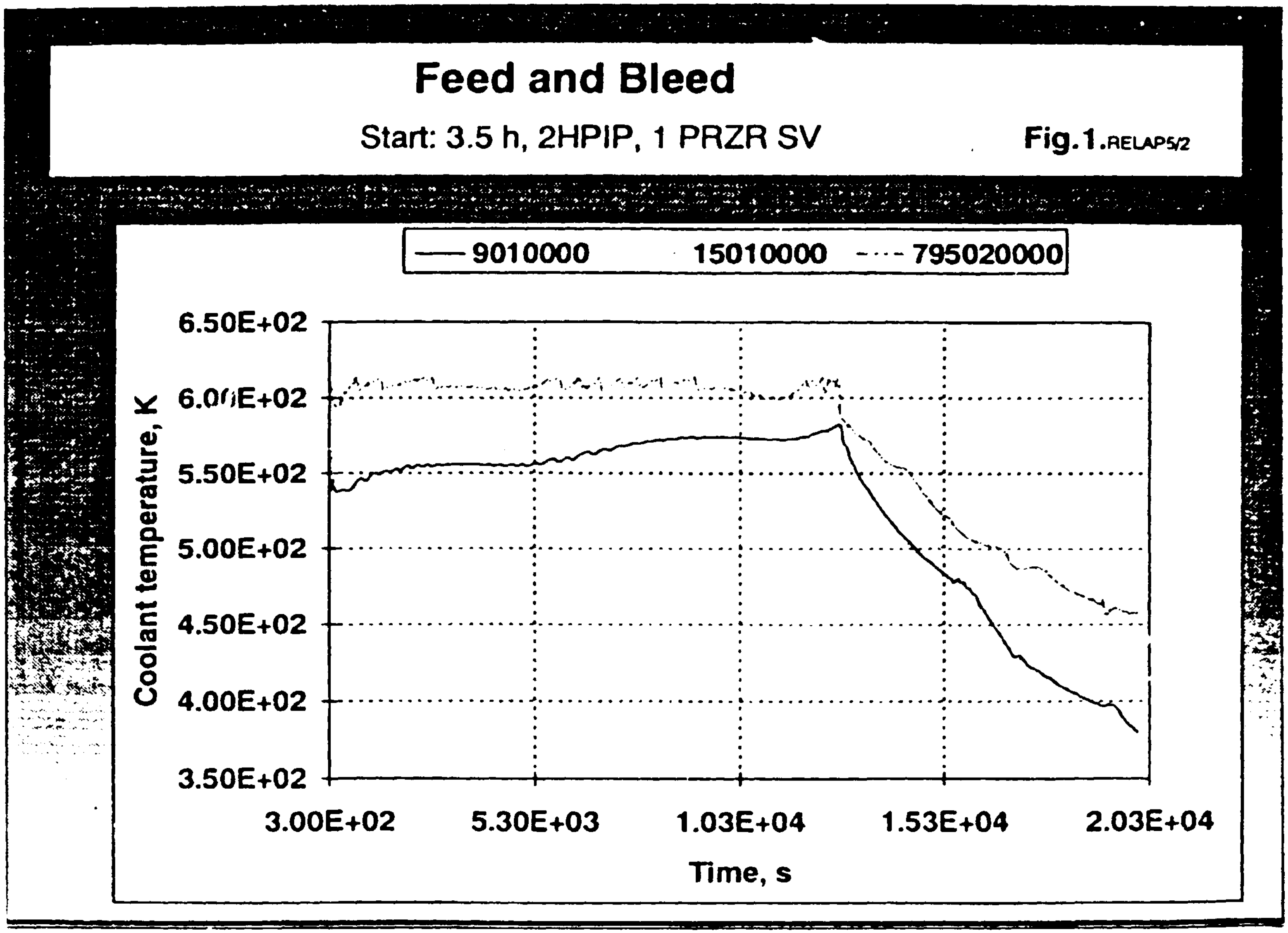
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Fig.1.RELAP5/2
9010000
15010000
795020000
Coolant temperature, K
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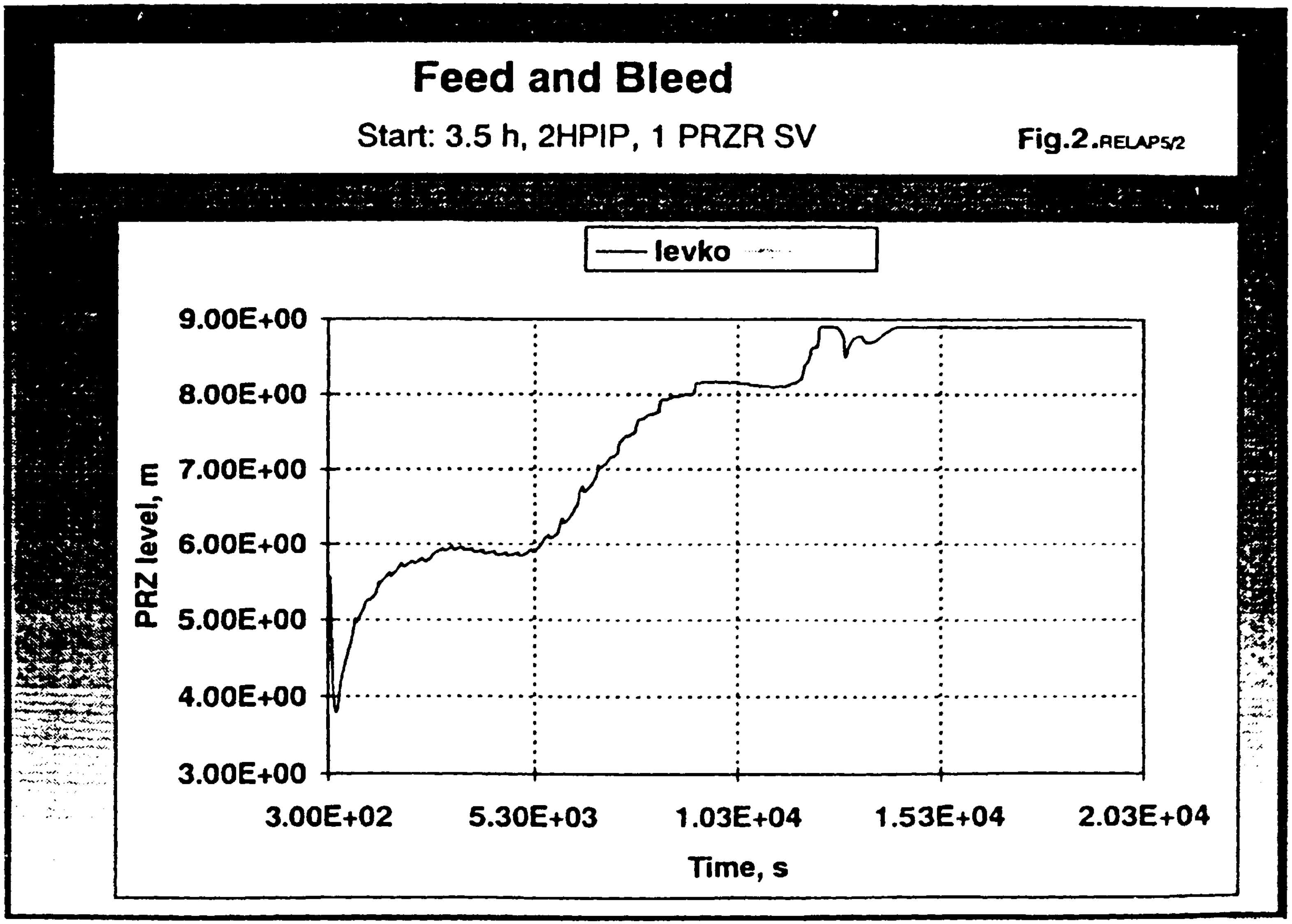
6.00E+02
5.50E+02
5.00E+02
4.50E+02
4.00E+02
3.50E+02
3.00E+02
5.30E+03
1.03E+04
1.53E+04
2.03E+04
Time, s



Feed and Bleed
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Fig.2.RELAP5/2
levko
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8.00E+00
7.00E+00
6.00E+00
5.00E+00
4.00E+00
3.00E+00
PRZ level, m
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5.30E+03
1.03E+04
1.53E+04
2.03E+04
Time, s

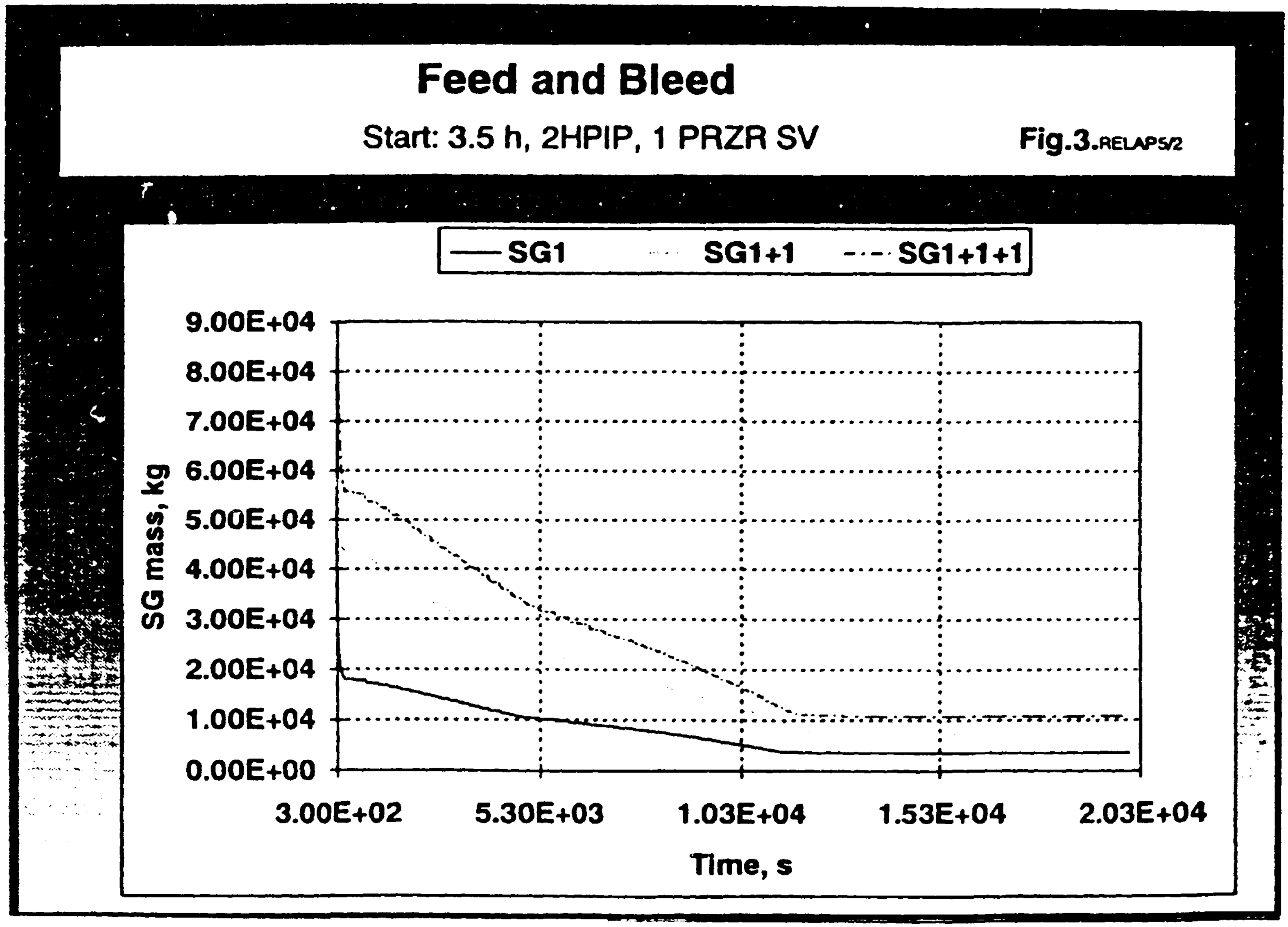
Feed and Bleed
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Fig.3.RELAP5/2
SG1
SG1+1
SG1+1+1
SG mass, kg
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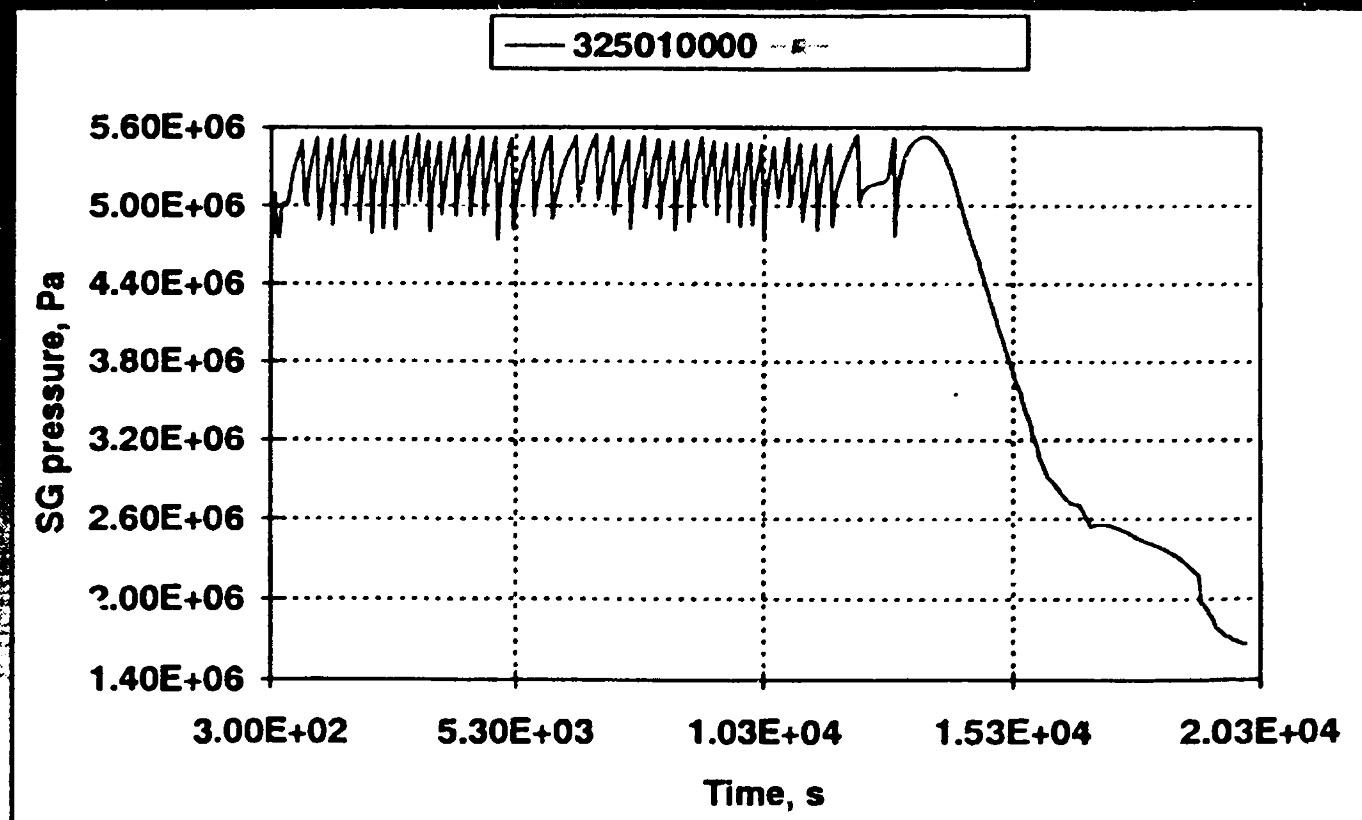
7.00E+04
6.00E+04
5.00E+04
4.00E+04
3.00E+04
2.00E+04
1.00E+04
0.00E+00
3.00E+02
5.30E+03
1.03E+04
1.53E+04
2.03E+04
Time, s

Feed and Bleed

Start: 3.5 h, 2HPIP, 1 PRZR SV

Fig.4.RELAP5/2



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WWER-1000(320) STEAM GENERATOR COLLECTOR RUPTURE. RADIOLOGICAL CONSEQUENCES

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Energoproekt Plc

One of the major acceptance criteria for design basis accidents at WWER types of units according to [1] is the non-violation of the intervention levels (actuation of the emergency plan of NPP) out of the exclusion zone, given in the following table:

Phase	Measure	Intervention level - dose (mSv)	
		Whole body (ore effective)	Single organ
Early (7 days)	Sheltering	5	50
	Stable Iodine	-	50 [a]
	Evacuation	50	500
Intermediate (1 year)	Control of food and water	5	50
	Relocation	50	

[a] Only for the thyroid gland

The table indicates only the lower limits of the intervention levels. The upper limits are by a power of 10 higher. The selection of an intervention level is done depending on the possibility to apply a certain measure.

From this point of view accidents with direct release of primary coolant to the atmosphere, bypassing the containment, are of a special concern. One of the possible routes for such a bypassing is any untightness of the heat-transfer structures of the steam generators through which the heat generated in the core is transferred to the secondary coolant. Taking into account the specifics of the steam generators PGV-1000M, most credible are the cases of partial or complete ruptures of SG tubes or of a lifting of the cover of a primary collector due to failure of the fixing bolts. The equivalent flow area of the break can vary from microscopic ruptures, through 2 x 1.37 cm^2 in case of a double sided rupture of an U-tube, up to 78.5 cm^2 in case of lifting of a collector cover. In the last case the limiting cross section is that of the annulus between the primary collector and the corresponding cover of the SG shell. The corresponding equivalent diameter is 100 mm. The last case is included in this report as having the strongest impact both on the primary circuit and core and on the releases to the environment.

The selection of the initial and boundary conditions for the analyses was done conservatively, so that to provide the most unfavourable conditions with respect to:

* heat removal from the core;
* heat removal from the primary circuit;
* heat removal from the secondary circuit;
* radioactive releases to the atmosphere.

The heat removal from the core is a function of the reactor power and coolant flow rate. The characteristic parameters are the DNB ratio, the difference between the saturation and coolant temperatures at the core exit and the maximum cladding temperature. Unfavourable effect on these parameters have the core peaking factors, the tripping of the reactor coolant pumps, the maximum delays of reactor scram and of the emergency core coolant systems as well as their minimal configuration. The earlier closing of the turbine stop valves results in a decreased heat removal from the primary circuit and, respectively from the core.

The heat removal from the primary circuit is a function of the primary coolant circulation, the status of the tube bundles of the steam generators and the availability of routs for heat evacuation from the secondary side. Unfavourable effect on these parameters have the earlier tripping of the reactor coolant pumps, the inoperability of the turbine by-pass system (BRU-K) and the earlier closing of the turbine stop valves.

The release of primary coolant to the atmosphere depends on the break flow rate and on the availability of BRU-K. The higher primary pressure, in general, leads to an increased break flow rate. From this point of view the operation of all three trains of the active part of ECCS has a negative effect on the radiological consequences of the accident.

The parameters that determine the source term are: the amount of primary coolant released to the atmosphere, the parameters of the release (height, enthalpy) and the initial activity of the primary coolant. The dose rates depend also on the atmospheric conditions at the NPP site and the restricted zone.

With respect to the above considerations, the following scenario was selected:

- The accident occurs during power operation at reactor power 102%, nominal primary pressure and coolant flow rate;

- The electrical heaters of the pressurizer are available before the loss of external grid, providing the pressurizer level is still high enough. The operation of the heaters decreases the primary pressure gradient and, respectively, leads to a higher break flow rate;

- It is assumed that, as a result of the turbine trip and the disconnection of the generator from the grid, a loss of external power supply occurs, leading to RCP trip and unavailability of BRU-K;

- The operators make no attempt to mitigate the consequences of the accidents during the first 30 minutes.

With respect to the controversial influence of the ECCS configuration, two boundary cases were analyzed: a minimum configuration of ECCS, as the most unfavourable with respect to core cooling and a maximum configuration of ECCS as the most unfavourable with respect to the radiological consequences.

The initial activity of the primary coolant was selected after comparing of operational data for the end of the third fuel campaign and data presented in the Technical design of Unit 5. As the values in the Technical design are higher by a power of 10 than those from the operation of the unit, the data from the Technical design were assumed for the analyses. To these, the activity of nuclides not included in the Technical design but measured at the plant were added. Some additional assumptions, leading to maximum activity of the released coolant, were also made in order to obtain conservative results:

- No aerosols are retained in the water volume of the affected steam generator. A justification for this assumption is the position of the break at the top part of the SG shell. Thus, the evacuation of aerosols with the steam is proportional to the steam release;

- Spiking effect S_a = 10 for the isotopes of Iodine and noble gases. This effect is due to the increased releases of the gases from the gap of the fuel rods through microdeffects during transients with pressure and temperature changes in the primary circuit. The spiking effect is assumed to be identical for all the isotopes of Iodine;

- Unfavourable meteorological conditions - atmospheric stability F (by Pasquille), wind speed 2 m/s and dry weather. An additional analysis was performed for the case of rain with intensity 5 mm/h.

After the start of the active part of ECCS, a dilution of the primary coolant occurs with respect to its activity. In order to obtain a more realistic estimation of the released activity, a special methodology was developed, calculating the actual amount of coolant with the initial activity in the primary circuit, which is released to the affected steam generator and thence to the atmosphere.

One of the main characteristics of the class of accidents with release of primary coolant outside of the containment, is the impossibility to restore the inventory of Borated water in the emergency tanks of ECCS. In this sense, the vapour void fraction of the coolant released through the steam dump to atmosphere facility (BRU-A) or SG safety valves is of major importance due to the fact that those are not qualified for a two-phase flow and so the possibility of a failure of an open valve to close should also be taken into account.

While selecting the most unfavourable scenario, the time and duration of the release to the atmosphere should also be accounted for. In this sense, the earlier release is more pessimistic due to the fact that during the first minutes of the accident the operators are nor expected to react adequately.

For the analyses actual data for the setpoints and capacities of the system for protection of the secondary side from overpressurization from Unit 5 of KNPP were used. For BRU-A these are:

Opening pressure	7.16 MPa (73 kgf/cm^2)
Closing pressure	6.276 MPa (64 kgf/cm^2)
Controlled pressure	6.67 MPa (68 kgf/cm^2)
Time for full opening/closing	15 s
Capacity of a fully open BRU-A at local steam line pressure 6.7 MPa	900 t/h

For the safety valves of the steam generators the corresponding values area:

	Opening pressure	Closing pressure
First SV	8.2 MPa (84 kgf/cm^2)	6.86 MPa (70 kgf/cm^2)
Second SV	8.4 MPa (86 kgf/cm^2)	6.86 MPa (70 kgf/cm^2)

Time for full opening:	1 s
Capacity of one valve:	800 t/h

With these setpoints and capacity of BRU-A, the opening setpoint of the safety valves of the affected SG is not reached. The maximum steam line pressure of the affected SG is a little above 8.0 MPa, i.e. nearly 0.2 MPa below the opening setpoint of the first SV. A combination of a higher opening pressure of BRU-A and a lower opening setpoint of the safety valves may result in an opening of a safety valve. Such a possibility exists in case of an earlier turbine trip or slower opening of BRU-A.

The performed parametric studies show that the control systems can not maintain the nominal parameters of the unit and in less than 10 s into the transient the reactor is tripped due to the decrease of primary pressure below 150 kgf/cm^2. The turbine trip is 1.5 s following the reactor scram and as a result the pressure in the steam generators rapidly increases. The first opening of BRU-A is 14 s into the transient, showing that the release of activity to the atmosphere can not be avoided.

The start of the active part of ECCS cools down the lower parts of the primary circuit, increasing the coolant density in them. The faster cool down in case of operation of 3 trains of ECCS leads to a faster decrease of the vapour void fraction in the core and to a decrease of the water level in the SG collectors, turning the break flow to two-phase. For this reason in this case the BRU-A is closed earlier. In a later phase the mass balance of the primary circuit turns positive, the primary pressure and, respectively the pressure in the affected steam generator increase. BRU-A is re-opened and a quasi-steady state of the primary circuit is established, the affected steam generator becoming part of it. The ECCS flow is equal to the BRU-A flow, the Boron solution injected to the primary circuit being released to the atmosphere. In case there is no operator intervention, the second opening of BRU-A is 18 min. into the transient. For its prevention the availability of a well justified emergency procedure for this class of accidents is of primary importance.

As a most unfavourable the case with radiological release to the atmosphere during the very initial phase of the accident should be considered. During this phase the operators have to evaluate the plant status and to diagnose the situation. In this sense the largest release occurs in case of a loss of external grid and availability of only one train of the active part of ECCS. In this case during the first 15 min. of the transient the total release of primary coolant to the atmosphere is 50.8 tons, of these 48.5 with the initial activity in the primary circuit. The release is 47 m above the ground level. The maximum dose rates are obtained at stable environmental conditions (Class F). The results are obtained assuming there is no evacuation or sheltering of people, with exposure duration 7 days. With this assumption the expected dose at the boundary of the restricted zone is 0.0182 Sv for the whole body and 0.184 Sv for the thyroid gland. The estimated dose rates (effective equivalent dose) on the site are 0.0521 Sv and 0.52 Sv for the whole body and the thyroid gland respectively. If Iodine prophylactics is applied, these values are decreased considerably. For example, if Iodine prophylactics is applied 6 hours after the initiating event, the dose rates are decreased twice, and in 3 hours - 4 times.

The stabilized dose rate at the boundary of the restricted zone at atmospheric stability F is approximately 3.10^{-4} Gy/h. At the NPP site the expected dose rate is $1.8 .10^{-3}$ Gy/h 2 hours after the accident. Later on it stabilizes and in 10-15 hours its value is 5.10^{-4} Gy/h.

In case of rain with intensity 5 mm/h the dose rates on the site are somewhat higher, 0.6 Sv and 0.2 Sv for the whole body and for the thyroid gland respectively. In this case the application of Iodine prophylactic and the evacuation of the personnel not engaged in the accident mitigation would be a reasonable measure leading to a decrease of the collective dose commitment.

As a result from the analyses, some main conclusions with respect to the evaluation of WWER-1000(320) safety can be drawn:

- The steam generator collector rupture accident is not threatening the heat removal from the core, even with a minimal configuration of ECCS. The maximum cladding temperature remains below its initial value.

- The heat removal from the primary circuit is accomplished both through the affected and the intact steam generators. Taking into account all possible inaccuracies in the adjustment of the opening setpoints of BRU-A and SG SV, the opening of a SG safety valve is possible. During the BRU-A operation the vapour void fraction of the coolant released through it is above 75%. Nevertheless, the possibility of a valve stuck open should be accounted for, especially is a safety valve is open.

- For the restoration of the mass balance of the primary circuit, the operation of one train of ECCS is sufficient. While selecting a strategy for accident mitigation, it should be kept in mind that primary coolant release is to outside of the containment, thus losing the possibility to restore the inventory of the emergency tanks. A procedure for filling of the tanks in emergencies should be also developed.

- The availability of a well justified emergency procedure for the class of primary to secondary leaks is very important for the prevention of severe consequences following the depletion of the safety injection tanks. Applying of such a procedure can also limit the primary coolant releases to the atmosphere within the first minutes of the accident, thus minimizing the consequences for the personnel and eliminating any consequences for the population.

- The radiological consequences of the accident are relatively small if an emergency procedure is applied after the 15-th min. of the transient. They do not require the applying of significant measures of the emergency plan if the weather is dry, such as evacuation of the population, because the analyses show that the intervention levels out of the restricted zone are not violated. A necessary measure is the evacuation from the site of any personnel not engaged in the emergency measures and in the operation of other units, while for the remaining in the restricted zone Iodine prophylactic should be applied during the first hours after the accident. Sheltering and/or Iodine prophylactic of the population in the wind direction may be applied in case of unfavourable atmospheric conditions. The probability that such measures may be needed is very low due to the very high conservatism, assumed in the analysis. The application of such measures should be after dose measurements on the territory outside of the restricted zone immediately after the radioactive cloud has passed.

- In case of unstable atmospheric conditions (A) and rain, the most unfavorable results for the NPP site are obtained. In this case the measures described above should be recommended. At the boundary of the restricted zone the loading is lower than in atmospheric stability of class F.

References:

1. IAEA Safety Series No 81 „Derived Intervention Levels for Application in Controlling Radiation Doses to the Public in the Event of a Nuclear Accident or Radiological Emergency", Vienna, 1986

BG9700033

METHODS, MEANS AND RESULTS FROM CONFINEMENT LEAKAGE DECREASE IN UNIT 2 AT KOZLODUY NPP

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The design concept of units 1-4 confinement at Kozloduy NPP does not comply to the state-of-the-art requirements contained in the standards for confinements in Western European nuclear power plants. Naturally, these requirements are stricter than those used in, the design of, these units, thus, making their direct application impossible. Kozloduy NPP reconstruction and upgrading, based on WANO's 6-month program, are aimed at finding a reasonable balance between the current safety requirements and the possibility to implement concrete technical solutions for confinement design improvement.

Within the framework of Theme D "Confinement Qualification" of the 6-month program, a long-term program to improve unit's 2 confinement characteristics was started with plans to transfer the experience and results to the other units with VVER-440 reactors as well. Work on this theme was a joint effort of engineers from Empresarios Agropados (EA), Spain and Energoproekt plc in close collaboration with staff from the reactor department of units 1-4 at Kozloduy NPP.

The following objectives were identified:

- Performance of experimental studies in order to assess confinement tightness (Local
- tests);
- Identification of ways and means to improve tightness;
- Performance of the most urgent repair works;
- Development of Global tests Procedures :
  - Leaktightness Tests
  - Structural resistance Tests
  - Flap Reliability Tests

In order to meet these objectives a method comprising Local and Global tests was developed.

A. LOCAL TESTS

I. COLLECTION AND SYNTHESIS OF INFORMATION ABOUT LEAKAGE PATHS IN CONFINEMENT

The potential leakage paths in confinement are all equipment which crosses the boundary or is by themselves boundary of the confinement, irrespective of their design and functions. They are classified in 9 groups, common for all four units, the only difference being the number of isolating valves of the suction air ducts to unit's and 4 ventilation systems B-2 and B-4.

The elements (potential leakage path) of all 9 groups for unit 2 are done below:

Group 1 **Heat, Ventilation and Air Conditioner(HVAC) Elements**: These, in turn, are grouped into 5 sections:

1.1.	Cut-off Valves	28 units
1.2.	Pressure Relief Valves	30 units
1.3.	System B-4 Swing Check Valves	1 unit
1.4.	System P-2 and B-2: Swing Check Valves	2 units
1.5.	System P-1 HVAC shafts	5 units

Group 2 **Doors and Suction Holes**

2.1	Doors (total number considered)	12 units
2.2.	Suction Holes (in System B-3)	10 units

Group 3	**Hatches and Manholes**		
	3.1	System P-1 Hatches	15 units
	3.2.	System P-1 Manholes	6 units
	3.3.	Steam Generator Hatches	6 units
	3.4.	Main Coolant Pump Hatches	6 units
	3.5.	Main Isolation Valve Hatches	2 units
	3.6.	Primary Water Purification System Hatches	7 units
	3.7.	Reactor Vessel Cover (Dome)	1 unit
	3.8.	Maintenance Hatch	1 unit
	3.9.	Gate between upper part of reactor shaft and transport corridor to the spent fuel pool	1 unit
Group 4	**Venting Flaps**		
	4.1.	Flap valve with diameter 520 mm	1
	4.2.	Flap valves with diameter 1130 mm	8
Group 5	**Pipe Penetrations**		
	5.1.	Used pipe penetrations	136
	5.2.	Reserve pipe penetrations	59
Group 6	**Valve Stem Extensions**		
	Total number: 131 units.		
Group 7	**Electrical Penetrations**		
	7.1.	type II with 8 penetrations	23
	7.2.	type III with 16 penetrations	21
	7.3.	type V with 4 penetrations	7
		TOTAL:	51

Group 8 Sump Drain Valves

The 6 units are located in the following rooms: A102. A004/3. A004/4. A010/A011. A013. B007(A012).

Group 9. Liner

The walls, floors and ceilings of the confinement rooms are lined by stainless steel sheets. The total length of the liner welds is approximately 12 km.

II. PERFORMANCE OF LOCAL TESTSII.

1 Test Methods and Procedures

Two methods - quantive and qualitive - have been used in the Local tests. For the purposes of applying them in tests of the individual penetration groups, Empresarios Agropados developed procedures stating the range of procedure application, test equipment, how to use the equipment, personnel qualification, instrument's calibration and the form of test result registration.

- QUANTITIVE METHODS - WITH MEASURING

⇒ Pressure Decay

82D-AT-Q-1002 REV.1: Pressure Decay Leakage Test Procedure. This document describe the methods used to determine penetration leak flow trough pressure drop measurement. It applies mainly to the HVAC valves located in ducts, May, 1992.

⇒ Flowrate Measurement

82D-AT-Q-1003 REV.1: Flow rate leakage Test Procedure. This document describes the techniques used to determine leak flow by direct measurement. It also applies to HVAC valves with an isolation function, May, 1992.

These Methods and Procedures are used for the "active" groups of penetrations [1].

- QUALITIVE METHODS - DETECTION

=> Vacuum Box

82D-AT-Q-1001 REV.1: Vacuum Box Leak Test Procedure. This procedure describes the method of testing welds and flat surfaces of the liner, May, 1992.

=> Ultrasonic detector

82D-AT-Q-1004 REV.1: Local Test Procedures Scope of Tests and Selection of Test Method. This procedure describes the different methods for detecting and evaluating the possible leak points in the penetrations of confinement, and defining the criteria to choose the pertinent method in every case ., July, 1992.

=> Blueling

82D-AT-Q-1004 REV.1: Local Test Procedures Scope of Tests and Selection of Test Method. This procedure describes the different methods for detecting and evaluating the possible leak points in the penetrations of confinement, and defining the criteria to choose the pertinent method in every case ., July, 1992.

=> Water Leak

82D-AT-Q-1004 REV.1: Local Test Procedures Scope of Tests and Selection of Test Method. This procedure describes the different methods for detecting and evaluating the possible leak points in the penetrations of confinement, and defining the criteria to choose the pertinent method in every case ., July, 1992.

=> Visual Inspection

82D-AT-Q-1004 REV.1: Local Test Procedures Scope of Tests and Selection of Test Method. This procedure describes the different methods for detecting and evaluating the possible leak points in the penetrations of confinement, and defining the criteria to choose the pertinent method in every case ., July, 1992.

These Methods and Procedures are used for the "passive" groups of penetrations and for material surface qualification [1] .

From tests of each group of penetrations a representative sample is made, considering which conclusions for the state of the whole group are drawn. From the tests of unit 2 at Kozloduy NPP such a sample was prepared for the electrical cable and the rod penetrations. All other groups, except for the pipe penetrations, whose testing was impossible, were 100% tested.

II.2. Test Results Form

Test results and measurements are entered special forms. A special columns show whether the measurement has been approved or repair of the tested item of equipment with prior tests and approval is planned.

In cases when sufficient amount of statistical information about the individual groups of penetrations has been gathered, repair permits without advance measurements can be also issued.

II.3. Results from Local Tests of unit 2

Penetration	Tested components, pcs	Method	Result
GROUP 1 - Penetrations for ventilation systems			
Cut-off valve system B-2	26/28	Flowrate leakage test	No leakage after repair
Pressure relief valve of system B-4	30/30	Pressure decay leakage tes	Leakage after repair
Swing check valve system B-4	1/1	UT	Leakage
Swing check valve system R-2	0/1	-	Unaccessible for measuring
Fan shafts system R-1	5/5	UT	No leakage after gland packing renewed
GROUP 2 - Doors and ventilation holes between doors			
Doors	12/13	UT + Blueling	No leakage after rubber seals replacement
Holes of B-3	0/10	-	Open system
GROUP 3 - Hermetic manholes			
Round manholes of R-1	6/6	UT	No leakage
Manholes	37/37	Water	No leakage after repair
GROUP 4 - Venting Flaps			
Venting Flaps	9/9	UT	No leakage
GROUP 5 - Pipe penetrations			
Pipe penetrations	0/131	-	Structure allows no testing
GROUP 6 - Valve stem extension			
Valve stem extension	25/131	Pressure decay leak test	No leakage after repair
GROUP 7 - Electrical cable penetrations			
Electrical cable penetrations	39/51 panels	UT	Sealed, no leakage
GROUP 8 - Drainage penetrations			
Sump drain valves	6/6	Water	No leakage
GROUP 9 - Liner			
Sheets	1-15%	Vacuum box	18% abnormalities
Welds	1-15%	Vacuum box	3% abnormalities**
Cladding	100	Visual inspection	42 remarks

Note: **Leakage from spent fuel pool. Holes without sealing Traces from oxidation

Local test results show the necessity of tests and repair during each refuelling cycle.

III. WAYS AND MEANS TO IMPROVE CONFINEMENT TIGHTNESS

Qualification of unit's 2 confinement showed that the ventilation system, penetrations are the major source of leakage to be expected because of their great number - more than 60 penetrations with size above 250 mm. Just for comparison, the designs of Western nuclear power plants impose a limitation of two 200 mm penetrations during normal operation. The isolating valves at the confinement boundary close up to 50% by gravity, and for the remaining 50% by electric drives with 30-60 s closing time. Isolating valves in the Western nuclear power plants are of the disk type with closing time 5 s.

One of the ways to upgrade confinement characteristic is to reconstruct and modify its ventilation systems. In /1/ are proposed the following modifications:

Repair/Modification Orders	**Activities**
	Ventilation system B-2
RMO-09	Reducing the number of suction valves from 14 to 9, and of the inflow valves from 14 to 11
RMO-11	Replacing the isolating valve dia 500 in suction air duct of B-2
	Ventilation systems П-4 and B-4
RMO-02	Reducing the number of excess pressure valves, dia 500 from 30 to 6
RMO-06	Installing a back-up disk valve in suction air duct of B-4 or replacing the present one
RMO-08	Making possible a local test of valve of RMO-06
RMO-117	Changing the type of drive of excess pressure valves dia 500, or replacing with better-quality valves
	Ventilation system B-3
RMO-08	Installing isolating valves, or removing the connections
	Ventilation system R-1
RMO-43	Replacing the sealing of fans' shaft by such providing the required tightness during operation.

Other ways to improve tightness are the periodic inspections and repair of the other groups of penetrations.

Group	**Means to improve tightness**
Doors	Planimetering
	Replacement of sealing
Manholes	Planimetering
	Replacement of sealing
Pipe penetrations	Chamber for periodic inspection of penetrations
Flap valves	Applying Greifswald NPP's experience
Electrical cable penetrations	Licensing of product for sealing of all penetrations

Speaking generally of the means for penetration's tightness, it should be noted that there is no experience in materials choice. In this respect, the experience at Bohunice NPP will be very useful.

B. GLOBAL TESTS

Empresarios Agropados and Energoproekt have developed procedures for global tests, namely:

82D-AT-Q-1005, Rev.1: Procedure for Global Confinement Tightness Test at reduced pressure for Kozloduy NPP units 1 to 4 , November, 1992

- Depressurization Stage to Verify Liner Leaktightness
- Pressurization Stage - 1.25 bar (abs.)
- Reduced Instrumentation to control the Test

82D-AT-Q-1006, Rev.1: Procedure for Global Confinement Tightness Test at nominal pressure for Kozloduy NPP units 1 to 4 , November, 1992

- Pressurization Stage - 1.6bar (abs.)
- Complete Instrumetation for Temperature and Humididy Correction

82D-AT-Q-1007, Rev.1: Procedure for Global Confinement Structural Integrity Test For Kozloduy NPP units 1 to 4 , October, 1992

- Pressurization stage - 2.15 bar (abs.)
- Pressure increments in gradual steps
- Cracks inspection and follow-up in singular points
- Displacements measurements versus maximum values by calculation

82D-AT-Q-1008, Rev.1: Venting Flaps Test Procedure for Kozloduy NPP units 1 to 4 , October, 1992

- Leaktightness test during closure and normal plant operation
- Calibration test on the Counterweight and the Stem Guide System
- Operating test on the Complete Assembly
- Complemented by Detailed Structural Analyses

In December 1992, a tightness test at pressure 1.3 bar (absolute) was carried out in unit 2 at Kozloduy NPP. After removing the untightness, mainly in the electrical cable penetrations, an equivalent leakage diameter of 103 mm was registered.

Many of the proposals relating to Item D, made at the end of the 6-month program, have been implemented at part of the units and the test results are indicative of the necessity and significance of this activity.

It is noteworthy, that in view of the future plans to build a filter-type ventilation system, the high degree of confinement tightness is one of the crucial design requirements.

REFERENCES:

1. Final Report for ITEM D, October 1992. Document of Impresarios Agropados 82D-FD-1005.

BG9700034

APPLICATION OF A TWO-PHASE INJECTOR IN THE SAFETY SYSTEMS OF NUCLEAR POWER PLANTS

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INTRODUCTION

The development of Safety Criteria for Nuclear Power Generation has caused the installation of complicated, multi-component Safety Systems.

In most of the currently operated PWR plants, the Active part of the Safety Systems (ASS) consists of Impeller Pumps (IP), Electric Motors (EM) and valves. The complicated layout (redundancy, cooling and lubrication) of these systems is obviously caused by the relatively low reliability of each of the basic components. On the other hand, the complicated layout influences directly the capital and production expenditures, finally rising the cost of generated electricity.

The main disadvantages of the IP - EM couple, which contribute to the high cost and complicated layout of the system are, as follows:

IP:
- physical incapability to deliver constant flow rate at varying head (hydraulic resistance);
- essential sealing;
- essential lubrication;
- essential cooling;
- includes moving (rotating) parts;

EM:
- essential electricity supply;
- essential lubrication;
- essential cooling;

The following article presents for discussion a concept for significant simplification, and consequently - cost reduction of Nuclear Power Plants ASS. The idea is to replace the IP-EM couple in some of the ASS with a two-phase injection jet device (IJD), without decreasing the overall ASS reliability. Considering the special physical properties of the IJD, such a replacement would eliminate the disadvantages listed above, although it will bring up new demands.

The authors aim to demonstrate the following:

- IJD is capable of fulfilling the functions of the IP-EM couple in certain ASS in PWR Nuclear Power Plants;

- The essential conditions for IJD functioning as a basic element of ASS are present in the current Nuclear Power Plants during all accidents, requiring activation of the relevant ASS.

A short review of IJD characteristics, revealed during our previous studies [1], is presented below.

I. INJECTOR JET DEVICE

The theoretical basis of the two-phase injector jet device (IJD) is presented in [2].

Newly discovered properties of this device are outlined in later works [3] and [4] which regard the IJD mainly as a means to increase the working medium pressure 1.5 - 2.0 times. This particular characteristic of the IJD, which is due to the unique processes occurring in its discharge sector, gives it a special advantage over all other jet devices.

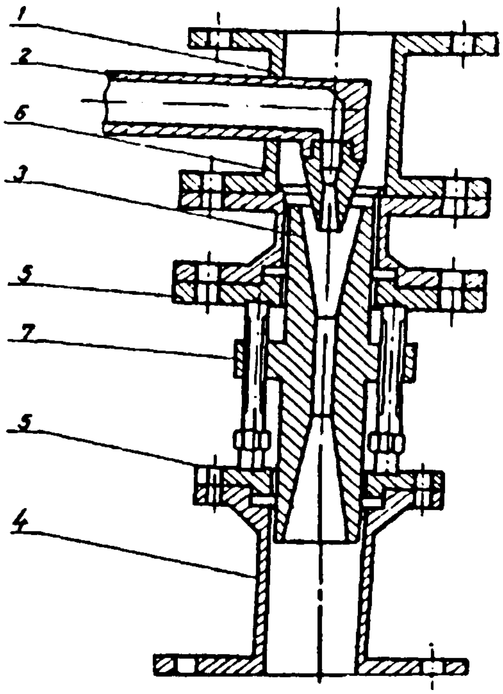

Fig.1

The principal structure of a two-phase jet injector is shown on fig.1. Physically, the IJD converts the kinetic energy and enthalpy of the working medium (steam) into potential in the condensation jump, which occurs in the mixing chamber. The working medium expands in the steam jet, producing low pressure, which draws the injected medium (water). The mixture flow is then slowed down in the mixing chamber, which causes pressure increase. Due to the working medium condensation and appropriately chosen geometry of the outflow part, it is possible to rise the outlet mixture pressure above the pressure of the working medium.

As illustrated on fig.1, the IJD consists of the following basic elements:

1 - injected medium (water) inlet channel;
2 - working medium (steam) inlet channel;
3 - injector working channel including inlet chamber, mixing chamber and diffuser;
4 - outflow (pumping) channel;
5 - fixing flanges;
6 - steam jet;
7 - working channel displacement mechanism.

The most interesting property of this device is the capability of sustaining constant flow rate regardless of the variations in the system's hydraulic resistance, outflow pressure being function only of the resistance, which is to be overcome. Thus, within certain limits, the IJD output pressure can vary between injected water initial pressure (P_b) and the maximal value (P_c) obtainable under the specific conditions (steam pressure, discharge flowrate), without operator intervention. A qualitative comparison between IP and IJD pumping heads versus flow rate is presented on fig. 2.

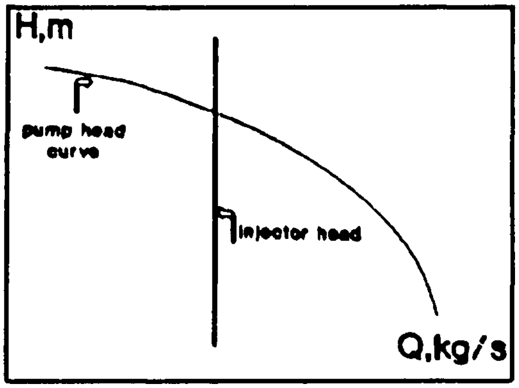

Fig.2

A method for engineering calculation of IJD properties is developed and validated against experiments, carried out with a sample IJD on a specially built test facility. Some of the results of this work are presented below.

1.1 IJD efficiency, steam and water flow rates.
In the theory of jet devices, productivity is assessed through the injection factor (U), which is the ratio between water and steam flow rates. The comparison between experimentally defined and calculated injection factors is presented on fig. 3 and fig. 4.

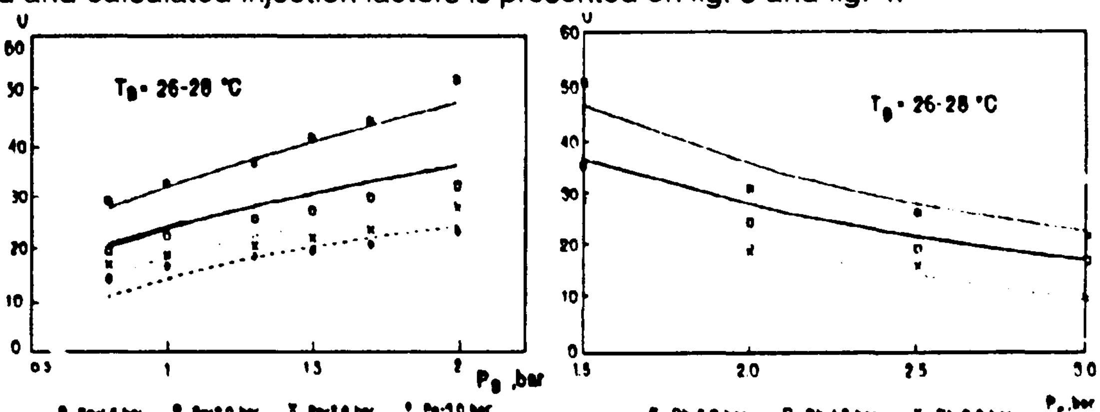

Fig.3

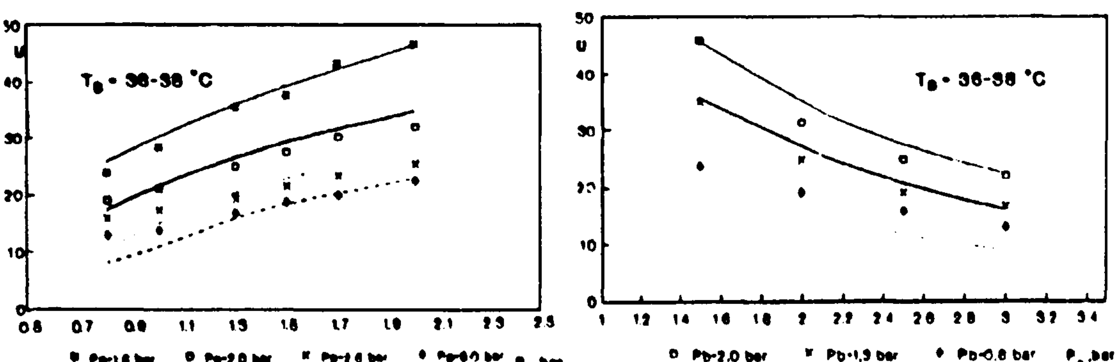

Fig.4

On all figures, the calculation results are shown with solid lines, and the experimental results - with data points.

Best coincidence is noticed at low pressures of the injected water, similar to the pressure in most ASS tanks. The results at varying temperature of the injected water (T_b) are of the same character. The overall relative error in defining of the working injection factor for all experimental values does not exceed 3%.

1.2 Maximal output pressure

The basic parameters, which influence the maximal IJD output pressure are presented on fig. 5. Especially interesting is the fact that the maximal output pressure remains practically constant while the injection factor changes significantly.

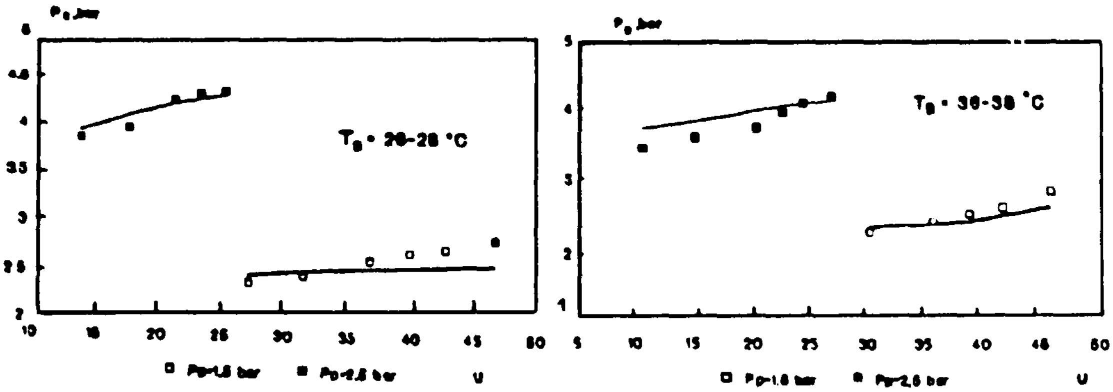

Fig. 5

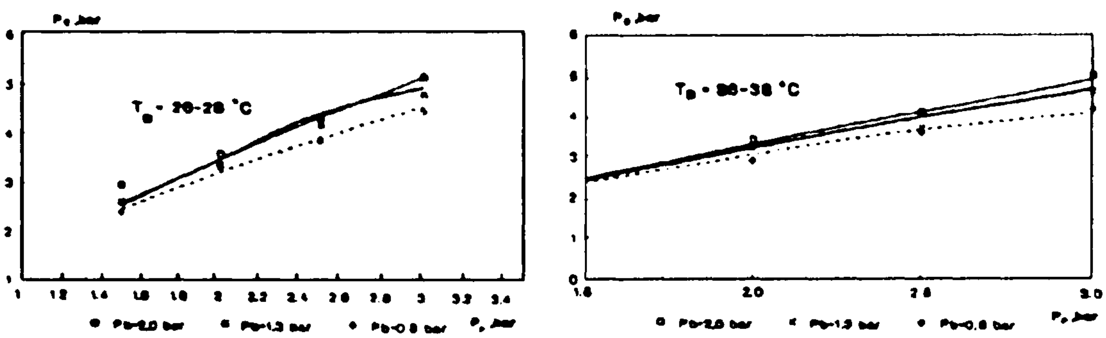

Fig. 6

The maximal output mixture pressure (P_c) is obtained in the experiments by adjusting the flow resistance of the test loop. Fig. 6 clearly shows that Pc depends mainly on the steam pressure (P_p), and for this particular IJD geometry $P_c/P_p \sim 1.7$.

There is a satisfactory coincidence between the experimental and calculated results. The maximal relative error does not exceed 10%.

Every point on fig. 5 and fig. 6 corresponds to the maximal output pressure for the relevant conditions. The complete flow - head diagram for each case is a straight vertical line (fig. 2).

The stability of IJD functioning improves when the inlet water temperature decreases.

1.3 Conditions for IJD implementation as a basic ASS element

1. Development of the necessary head

The head developed by IJD depends only on the working medium (steam) pressure. The geometry of the IJD presented here allows the development of a head which is 1.7 times greater than the working medium pressure.

2. Development of the necessary flow rate

The constant flow rate, sustained by the IJD during the whole transient allows the whole range of necessary heads to be covered by only one type of IJD. The necessary flow rate can be developed either by a larger capacity IJD or by multiple IJDs of a smaller capacity.

3. Reliable supply of working medium

Steam Generators (SG) are regarded as suitable sources of working medium for accidents with decrease of the primary side pressure, and the Pressurizer (PRZ) for accidents with increase of the primary side pressure. The availability of more than one SG rises the problem of optimal feed line layout. The delivery of working medium with certain parameters, necessary for different accidents will be discussed below.

4. Temperature of the injected water

When the difference between the temperatures of the working medium (steam) and the injected medium (water) decreases, IJD productivity decreases too. In some accidents this will impose the necessity for cooling of the injected medium. This necessity may limit the ASS independence from electricity supply, depending on the precisely evaluated possibility for passive cooling of the injected medium.

II. APPLICATION OF IJD IN THE ACTIVE SAFETY SYSTEMS (ASS)

In this section we shall comment briefly on some basic accidents in Nuclear Power Plants with PWR reactors. These accidents are studied, modelled and described in great detail in many previous studies [5,6].

While commenting on the accident scenarios, our point of concern will be the ASS action, necessary to bring the reactor to a safe state. From this point of view, the vast variety of possible accidents may be split in two basic types:

a) accidents without depressurization of the primary side, where the major ASS function is to supply the SG emergency feedwater system;

b) accidents with depressurization of the primary side, which require activation of the Emergency Core Cooling System (ECCS).

We shall show, that ASS functioning does not deteriorate if the IP-EM couple is replaced with a IJD, where such a replacement is possible. We shall also show, that the conditions for IJD functioning as an element of the ASS are present during the considered accidents. Finally, commenting on the analysis of one of the most probable severe accidents, we shall demonstrate the importance of one of the IJD main advantages - the independence from electricity supply.

II.1 Accidents without primary side depressurization

Regardless of the variety of initial causes and scenarios, all these accidents set the same problem for ASS action: feed water must be delivered to the SGs with such parameters and at such flow rate, that safe removal of the residual core heat is guaranteed for long periods even if only one SG is available.

In order to meet this demand, the IP-EM couple performs two functions:

a) delivery of the necessary flow rate;

b) development of the head, necessary to overcome the pressure difference between the source (Deaerator, tanks) and consumer (SG) of feed water, including the hydraulic resistance of the relevant line;

Let us now consider the capability of the IJD to perform these functions, using as a working medium the steam from intact Sgs, as shown on fig. 7.

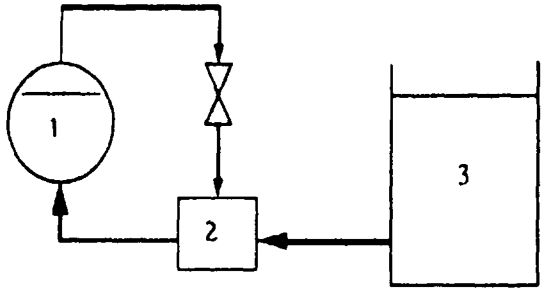

Figure 7.
1-Steam Generator 2 - Injector (IJD) 3 - Water Storage Tank

II.1.1 Flowrate of the emergency feedwater supplied by IJD

The delivery of the necessary flowrate depends on four factors:

- IJD geometry;
- working medium (steam) flow rate;
- water inventory in the source;
- temperature difference between the working medium and injected medium (water).

If the other three factors are present, the geometry of the IJD described in Chapter I provides a constant flow rate. As it was mentioned above, this flowrate can be easily varied by installation of multiple IJDs.

The necessary flow rate can be assessed by the injection factor, which is the ratio between water and steam flow rates. For the studied IJD the injection factor is within the range 10 - 30. Obviously, the steam produced even by one SG will be sufficient for a IJD with such injection factor to feed all (3, 4 or 6) Sgs.

The water inventory in the storage tank is not influenced by the replacement of the IP-EM with a IJD.

The necessary temperature difference between the working medium and the injected medium is certain to be provided during the whole transient, because of the different conditions in the SG and in the water storage tank.

It should be noted, that the reliability of steam supply does not depend on steam sources (their capacity being many times larger than the necessary), but on the feed lines, valves and controls. This problem should be addressed in detail during the reliability assessment of the emergency feed water system as a whole.

II.1.2 IJD head for emergency feed water supply to Sgs

Having in mind, that the pressure in the water source is close to atmospheric, and that the hydraulic resistance of the lines does not exceed 3% of the SG secondary pressure (P_{sg}), we conclude, that IJD must develop a head of ~ 1.05 P_{sg} in order to provide feed water to the relevant SG during the accident.

In Chapter I. we have already shown that IJD output pressure depends linearly on working medium pressure, with a factor of 1.5 - 1.7 for this particular IJD. Obviously, the SG being a source of working medium and consumer of the injected medium in the same time, the necessary head is always available with a considerable margin.

II. 2 Accidents with depressurization of the primary side

We shall focus our attention mainly on the accidents with leakage of primary coolant into air-tight compartments. In this case, the major part in accident mitigation is played by the ECCS pumps. The replacement of the IP-EM to IJD raises similar problems, most important of which are the following:

- development of the head necessary to overcome the pressure in the primary side and the hydraulic resistance of the line;
- sustaining of the temperature difference between the working medium and the injected medium.

Here we assume that IJD is supplied with steam from SG(s), as shown in fig.7.

II.2.1 Development of the necessary head

Considering the behaviour of primary pressure, we can split these accidents in two types: accidents with large loss of coolant (LB LOCA) and accidents with small loss of coolant (SB LOCA). The main difference between them is, that in case of LB LOCA the primary side empties quickly - within 30 seconds, while in case of a SB LOCA primary pressure is kept quite high (approximately equal to SG secondary pressure) for a relatively long time.

During the analysis of LB LOCA [7], no attention was paid to the pressure in the Sgs. This is due to several reasons, one being the imperfect computer codes, used for accident modelling. Obviously, in such accidents, the SG steam can not be used as IJD working medium due to the low (if any) heat transfer to the secondary side. Further analysis is necessary in order to prove IJD capability to cope with such accidents, using other sources of working medium. One possible solution is to use as working medium the steam, generated in the core during the accident.

In this report we shall focus only on SB LOCA accidents.

The IJD presented here can be used in such accidents, being capable of increasing the output pressure to 1.7 times the working medium source pressure. For PWR reactors of the VVER-1000 type, the IJD can start working on SG supplied steam when primary pressure has decreased to ~ 10 MPa. In a large number of SB LOCA models studied by the authors this pressure occurs during the first 10s of the transient.

When a SB LOCA occurs together with a loss of the NPP internal electricity supply, IJD could start delivering water to the primary side much faster than the standard ECCS pumps, which depend on Diesel Generators (DG) start up time.

II.2.2 Sustaining of the temperature difference between the working medium and the injected medium

The reliable functioning of IJD requires a certain temperature difference between the working medium and the injected medium. This temperature difference influences considerably IJD head, bringing it to zero if reduced below certain limit.

After its reserve water inventory is exhausted, the ECCS switches to recirculation from the reactor building sump, thus bringing the leaked (and still hot) water to the primary circuit. At this stage the sump contains saturated liquid, and reactor building safeguards are activated to reduce the pressure in the relevant compartments. Thus, the sump water temperature is hardly above 100 C. On the other hand, the steam pressure in the SG is determined by the functioning of SG steam dump valves and safety relief valves, which keep it high enough to provide the necessary temperature difference. Hence, the introduction of an IJD provides better conditions for the functioning of ECCS heat exchangers and reduces the risk of pressurized thermal shock in the reactor vessel, being capable of pumping warmer water than the IP-EM couple during sump recirculation.

II.3 Beyond Design Basis Accident : Complete blackout with failure of the Diesel Generators to start

Beyond Design Basis Accidents have very low probability of occurring, but nevertheless are seriously analyzed, especially after the Three Mile Island and Chernobil accidents.

"Complete blackout" means loss of electricity supply to all consumers, except those of category I. Being dependent on electric power, the IP-EM couple is inoperable in such conditions, thus making the whole ECCS unavailable. The analyses of this scenario for VVER-1000 [8] show, that core uncovering occurs at ~ 242 minutes after the beginning of the transient, due to primary coolant blow-out through PRZ relief valves. The smaller water inventory in the primary side of western (e.g. USA) PWRs makes this period 2 - 4 times smaller.

The replacement of IP-EM to IJD in this case eliminates the necessity of electricity supply, thus making the ECCS available and bringing the accident into the range of accidents without primary side depressurization - foreseen and manageable by design. Here we come to the conclusion, that IJD introduction in ASS eliminates one of the most possible severe accidents in PWR nuclear power plants.

CONCLUSIONS

The short analysis of different typical accident regimes shows, that the introduction of IJD in the Active Safety Systems of Nuclear Power Plants brings significant advantages in comparison with the currently used IP-EM couple. The easiest to apply and the most advantageous is the introduction of IJD in the SG emergency feed water system. Such introduction will make this system completely immune to loss-of-electricity supply accidents.

The IJD based feedwater system can be connected to the primary side as well. Thus one and the same system could be used both for emergency feedwater supply to the Sgs or for emergency core cooling, depending on the type of the accident.

The introduction of IJD in the current ECCS is now limited to accidents with small loss of reactor coolant. Additional studies are necessary in order to evaluate the possibility for coping with the whole range of possible leaks and to find other sources of working medium at suitable conditions.

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НЕРАЗРУШАЮЩИЙ КОНТРОЛЬ В АЭС "КОЗЛОДУЙ"

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Объеспечение безопасной эксплуатации сооружений и систем в АЭС связано с внедрением и развитием методов неразрушающего контроля (НК). Методы НК начали применяться в АЭС "Козлодуй" до пуска станции. В 1973 г. сформирована первая группа по неразрушающему контролю для проведения входящего контроля сооружений и доэксплуатационного контроля систем в условиях наладки оборудования. В 1974 г. сформирована лаборатория "Металлография и герметичност" в составе которой включена и группа по НК. С 1991 г. в АЭС "Козлодуй" работает отдел "Диагностика и контроль", которой является самым развитым центром в Болгарии для проведения комплексного (разрушающего и неразрушающего) контроля и диагностики высокоответствен-ного энергийного оборудования в процессе эксплуатации.

Отдел "Диагностика и контроль" осуществляет активные международные контакты со специалистами Нововоронежкой, Запорожкой и Балаковской АЭС, с лабораториями заводов производителей оборудования в Ижорске и Подолске, с научно-исследовательскими институтами ЦНИИТМАШ - Москва, Гидропресс и др. При эксплуатации АЭС "Козлодуй" полезными являются связи с ведущими институтами в Болгарии - с Институтом механики и Институтом металловедения Болгарской академии наук, с СИМЕ КОНТРОЛЬ, с Техническим университетом в Софии и с Техническим университетом в Варне, а также с институтами и фирмами с утвержденными традициями в диагностике атомных электростанций INETEC Загреб, TECHNATOM - Испания, ZETEC - США, Westinghause Instrument США, Force Institute Дания и др. В отделе "Диагностики и контроля" работают свыше 60 специалистов I, II и III уровня по неразрушающим методам контроля, подготовленных в ВУЦ "Квалима" в соответствии с требованием EN 473. Ведующие специалисты II и III степени получили и дополнительную квалификацию в учебных центрах вышепредставленных институтов в области автоматизации оптического, вихретокового и ултразвукового контроля сооружений АЭС.

Отдел "Диагностика и контроль", в сотрудничестве с научными, научно-внедрителскими институтами и международными организациами (VANO-TAT), выполняет следующие основные задачи в АЭС "Козлодуй":

- Техническое обслуживание оборудования в условиях входящего и эксплуатационного контроля. Контроль в условиях аварий.

- Разработка нормативных документов, регламентирующих контроля и диагностики методики, технологии, заводские стандарты, технологические инструкции.

- Разработка технических средств контроля.

- Лабораторный контроль материалов.

- Разработка рекомендаций и экспертных оценок с целью обслуживания организаций, обслуживающих АЭС "Козлодуй".

- Разработка база данных технических характеристик материалов, оборудования и регистрированных изменений с целью определения ресурса энергетического оборудования.

Неразрушающий контроль в АЭС "Козлодуй" осуществляется при помощи всех видов контроля (оптического, проникающих жидкостей, радиационного, ультразвукового, вихретокового, герметичности, магнитного, металлографического, спектрального и нискочастотного акустического). Использованное оборудование для контроля позволяет, при помощи автоматизированных и компьютеризированных систем, исследование, формирование базы данных и дополнительный анализ на основе специлизированных программных продуктов.

В соответствии с требованиями действующих стандартизационных документов для объеспечения безопасности АЭС, за последные годы разработаны нормативные и технологические документы, регламентирующие контроля в АЭС. Реализована хорошая гармонизация методов, средств и метрологического объеспечения контроля с существующими требованиями на этапе строителства (документы ПК 1514-89, ОСТ 108.004.108-80 [1, 3]) и эксплуатации (документы ASME [3]). Ниже представлен список в котором включены основные нормативные документы по НК в АЭС.

1. Программа развития эксплуатационной технической диагностики АЭС "Козлодуй".

2. Организация работы отдела "Диагностики и контроля".

3. Технологическая инструкция для контроля методом проникающих жидкостей.

4. Технологические документы автоматизированного ултразвукового неразрушающего контроля несплошностей в сварных соединениях трубопроводов ДУ 500 и ДУ 200 ВВЕР-440.

5. Технологические документы по применению радиационного неразрушающего контроля.

6. Технологические документы по определению ферритной фазы в сварных соединениях из аустенитной стали.

7. Технологические документы по визуальному контролю ДУ и элементов АЭС.

В работе представлены, прежде всего, технологические документы 1 и 4 . Целью программы развития эксплутационной технической диагностики является повышение безопасности и надежности АЭС на основе: обнаружения недопустимых отклонений в свойствах и сплошности материалов, прогнозирования ресурса сооружений, улучшения эффективности эксплуатации в условиях оптимальной нагрузки оборудования, уменьшения продолжительности ремонта, сочетания плановых ремонтов с современными системами диагностики, сочетания принципов эксплуатационного обслуживания по ресурсу с эксплуатацей по техническому состоянию, использования возможностей предупредительной диагностики и увеличения объема ремонтных работ, связанных с оборудованием, с отклонениями в эксплуатационных харак теристиках.

Программа предусматривает сочетание периодического контроля с непрерывным, которой применяется практически к всем подлежащим контролю соединений и сооружений. Основные направления развития непрерывного контроля и диагностики в АЭС связаны с решением следующих задач:

1. Техническая диагностика топломеханического оборудования (контроль металла, контроль механических напряжений, регистрация основных параметров рабочего режима, механо-математическое моделирование объектов контроля, формирование классификационных критериев, оценка ресурса).

2. Техническая диагностика электрического оборудования (контроль, моделирование объектов, формирование классификационных критериев, определение ресурса).

3. Развитие методов генезиса, диагностики и прочности материалов и сооружений.

При выполнении программы особое внимание уделено:

- вводу систем компьютерного архивирования результатов контроля и экспертных оценок;

- метрологическому объеспечению контроля с целью повышения точности и надежности контроля;

- вводу эксплуатационных систем многопараметрого неразрушающего контроля, оптимизацию контроля на основе выбора новых информационных параметров и применения новых методов и средств;

- развитию методов механо-математического и физического моделирования состояния материалов;

- созданию неразрушающих критериев эксплуатационной прочности, остаточного ресурса и характеристик риска;

- вводу системы супервейзерского контроля;

- созданию единной системы подготовки и переподготовки кадров, а также эффективной оценки их подготовки в соответствии с требованиями EN 473;

- созданию учебно методического центра подготовки кадров по неразрушающему контролю в АЭС.

Разработанная программа выполняется отделом "Диагностики и контроля" АЭС "Козлодуй" в сотрудничестве с ведущими институтами в Болгарии в области неразрушающего контроля ИМех БАН, СИМЕ - КОНТОЛЬ, ТУ - София.

В качестве примера подготовки технологической документации представлены результаты исследования и разработанные документы по ультразвуковому неразрушающему контролю сварных соединений из аустенитной стали в трубопроводах ДУ 500 ВВЕР-440, реализированного при помощи механизированной компьютерной ультразвуковой системы P-scan (Дания)[4]. Проведены комплексные лабораторные исследования характеристик металла трубопровода и сварных соединений. На рис.1- рис.3 представлены микроструктура исследованных сварных соединений, распределение скоростей распространения продольных C_L, вертикально поляризированных поперечных C_{tV} и поверхностных C_r волн и распределение коэффициентов затухания продолных α_L и поперечных α_t волн в зависимости от угла θ между ориентацией дендритной структуры и направлением распространения ультразвуковых волн. На рис.4. показано распределение ферритной фазы на поверхностях А и В сварного соединения. Плотные и пунктирные линии использованны для представления резултатов контроля соединений, выполненных ручным и автоматическом способом сварки.

Анизотропия структурных, акустических и магнитных свойств материала сварного шва требует разработки новой методики формирования акустического тракта ультразвукового контроля. Для случая обнаружения несплошности Н в сварном соединении (рис.5) при помощи наклонного щупа О акустический тракт записивается в виде [4]

$$A_0' = A_0\left[\prod_{i=1}^{4} e^{-2\alpha_i L_i}\right]\left[\prod_{i=1}^{3} D_{i,i+1} D_{i+1,i}\right] F_H Q_A(\varphi) R_{OB}$$

где A_0 и A_0' амплитуды генерированного и принятого ехо-сигнала α_i коэффициент затухания ультразвуковых волн; L_i акустический путь; $D_{i,\,i+1}$ амплитудный коэффициент прохождения на границе сред i, и i+1; F_H- функция, учитывающая влияния характеристик направлености наклонного щупа; $Q_A(\varphi)$ функция, учитывающая отражающие свойства несплошности, R_{OB} коэффициент отражения на границе основной метал-воздух; i индекс среды (1, 2, 3, 4 соответственно, для материала наклонного щупа, для основного металла и для различно ориентированных областей сварного соединения).

На основе результатов лабораторных и теоретических исследований по оценки степени прохождения ультразвуковых волн через границу между двумя средами при наличии или отсуствия граничного слоя, по исследованию отражения ультразвуковых волн от контрольных несплошностей, по оптимизацию акустического контакта, по оценки отношения полезного ультразвукового сигнала от несплошности к структуным шумам в анизотропном материале сварного соединения, предложены нормативные технологические документы, которые регламентируют условия проведения исследований и неразрушающего контроля сварных соединений в трубопроводах ДУ 500. Основные требования по оценки применимости и по регламентированию условий контроля отражены в следующих документах.

1. УКДУ1. Технические условия дефектоскопичности сварных соединений при ултразвуковом контроле (УЗК) трубопроводов ДУ 500.

2. УКДУ2 Правила оценки основных характеристик сварных соединений при УЗК ДУ 500 ВВЕР-440.

В разработанных технологиях УЗК различных типов сварных соединений в ДУ 500 обоснованы режимы настройки чувствительности контроля, сделан выбор информационных параметров для охарактеризирования несплошностей и обоснована система оценки дефектности. В соответствии с требованиями ASME [2] для настройки чувствительности и допустимости обнаруженных несплошностей по амплитуде отраженного сигнала выбран контрольный цилиндрический отражатель диаметром 3.2 mm. При схемах контроля в которых предусматривается изменение расстояния от щупа до несплошностей от 30 до 135 mm рекомендован эквивалентный цилиндрический плоскодонный отражатель с диаметром от 2.6 до 3.9 mm [5]. Настройка ультразвукового устройства реализируется с учетом методик построения дистанционно-амплитудных коректирующих кривых (ДАК) [4]. Режим регистрации с 6 dB выше режима браковки. Режим контроля, с учетом структурных особеностей материала выбран с 14 dB выше режима браковки. Норма браковки регистрированных единичных несплошностей выбрана по ПК 1514-89 [4].

Для метрологического объеспечения контроля разработана система сравнительных образцов и контрольных блоков. Контрольный блок ОКБ4-А предназначен для оценки характеристик наклонных преобразувателей. Контрольный блок БКБГ-А соответствует регламентированному ASME базисному калибрационному блоку. Дополнительно, для каждого вида сварного соединения, созданы сравнительные образци для настройки преобразувателей, для построения ДАК кривых, для оценки допустимых несплошностей.

Основные схеми прозвучивания сварных соединении ДУ 500 представлены на рис. 6 для случая поперечных (а), продольных б) и угловых (в и г) соединений [6]. Основная схема прозвучивания реализируется при помощи прямого луча преобразувателем типа ДУЕТ с углами преломления 70^0 и углом сходимости 14^0 (рис. 6. а, в, г). Описаные схемы, при соотношении полезного сигнала к шуму 12-14 dB, позволяет уверенного контроля зоны корня сварного соединения. Прозвучивание при углах 70^0 гарантирует уменьшение амплитуд ложных сигналов от нижней части сварного шва. Основная схема прозвучивания продольных сварных швов реализирована при помощи наклонного преобразувателя продольных волн [6]. При угловых сварных соединениях, из за трансформации отраженииях продольных волн от плоских граничных поверхностей, наблюдаемая картина на экране ультразвукового устройства сложна для анализа. При помощи стандартных наклонных преобразувателей проверяется только небольшая часть соединения (зона термического влияния и 5-8 mm материала сварного шва). Предусмотрено и применение дублирующих схем типа ТАНДЕМ для контроля корня сварного шва и зоны термического влияния (рис. 6 а, б, г). Применяемые схемы гарантируют уверенное обнаружение опасных эксплоатационных несплошностей типа трещин. Зона корня прозвучивается при помощи бинарного наклонного симметричного преобразувателя LLL $60^0/60^0$, а приповерхностная зона с преобразувателями для ассимметричного тандема LLT 70L/17T и 70L/30T.

На рис. 7 и рис. 8 представлены некоторые результаты, полученные при настройки наклонных преобразувателей VSY60 и MWB45-2 с применением контрольного блока БКБГ- А. Разработанные схемы контроля обеспечивают оптимизация УЗК и уменшения радиационного воздействия.Они адаптирсванных к механизированному комплексу P-scan.

Разработанные технологические документы контроля сварных соединений ДУ 500 ВВЕР-440 согласованны с ИБИАЭ КИАЭМЦ и внедрены в практике отдела "Диагностики и контроля" АЭС "Козлодуй".

Подготовка доклада осуществлено при помощи спонсорства по проекту ТН 317 НФНИ.

1. ПК 1514-89 "Правила контроля сварных соединений и наплавки узлов и конструкций АЭС, опытных и исследовтельских ядерных реакторов и установок", М., 1990.
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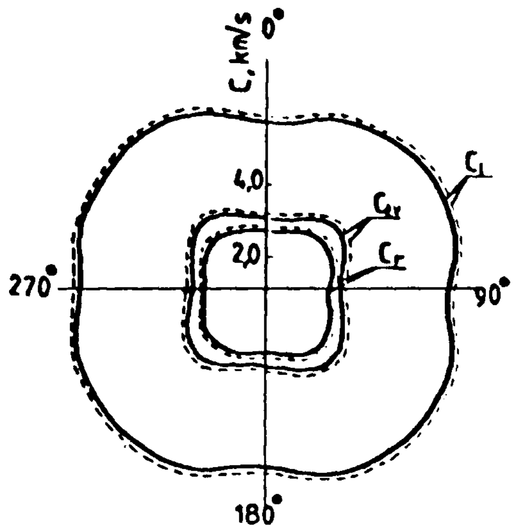
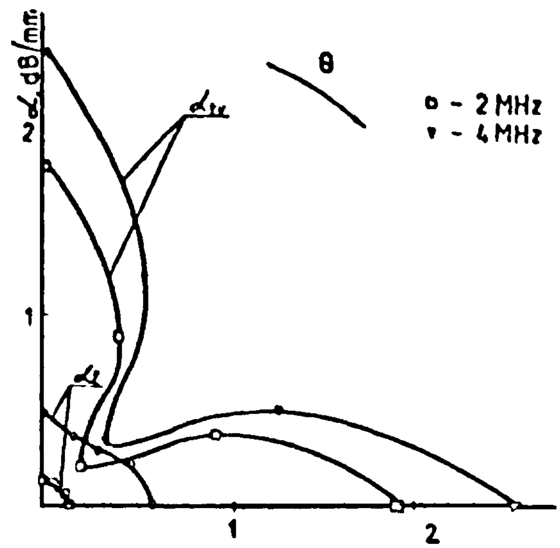
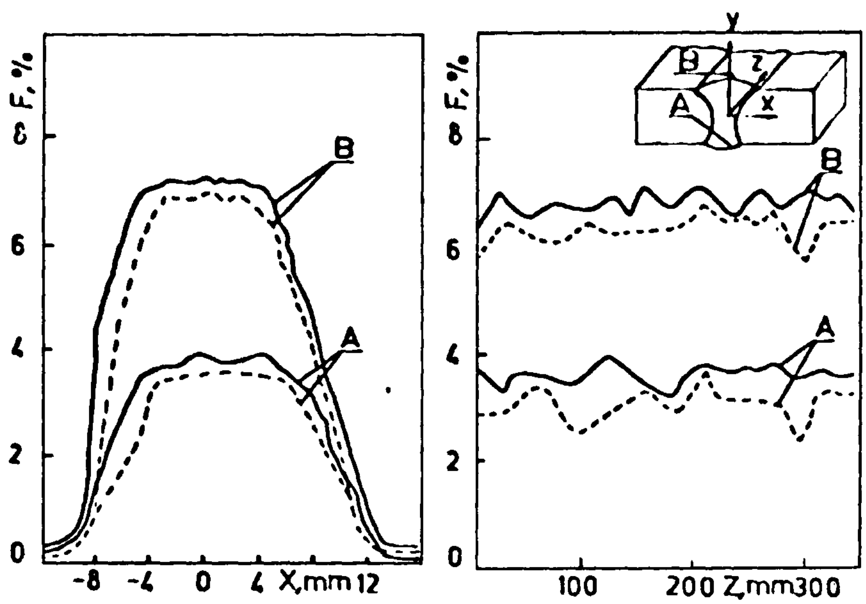
фиг. 1



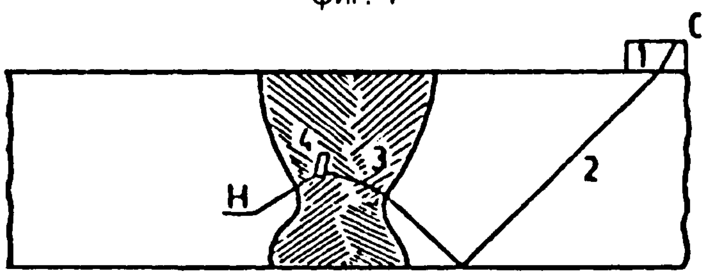
фиг. 2



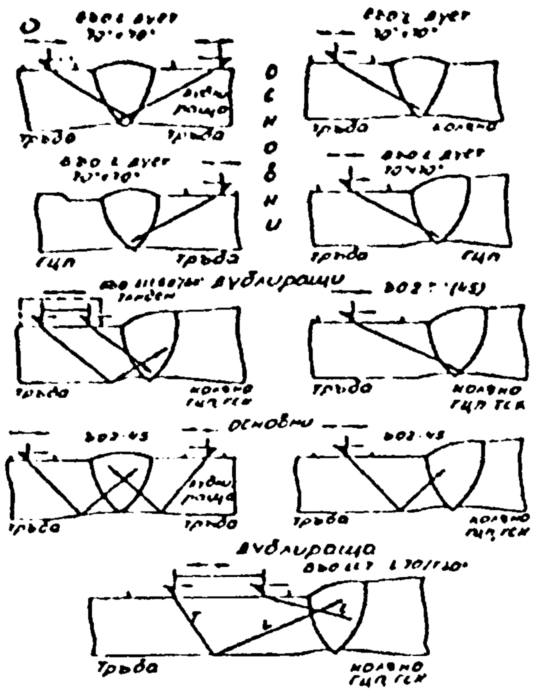
фиг. 3



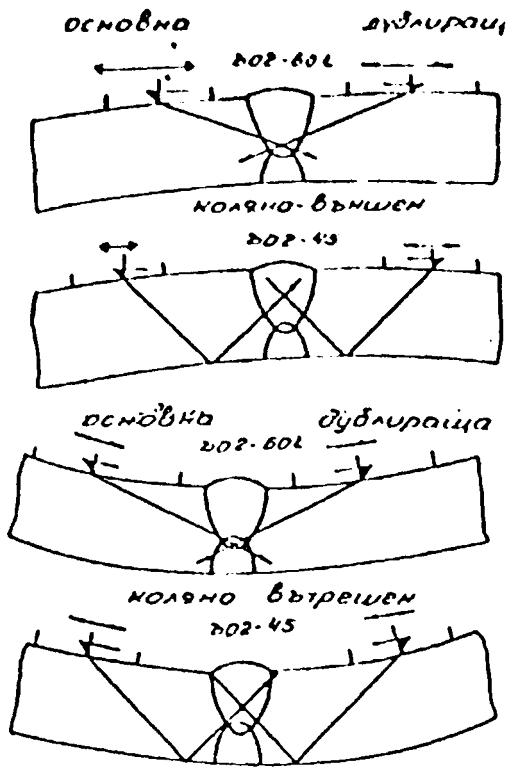
фиг. 4



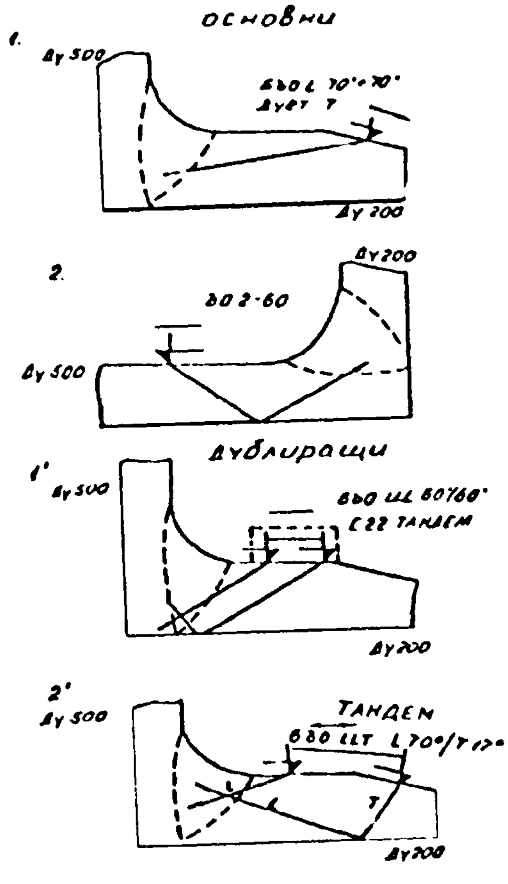
фиг. 5



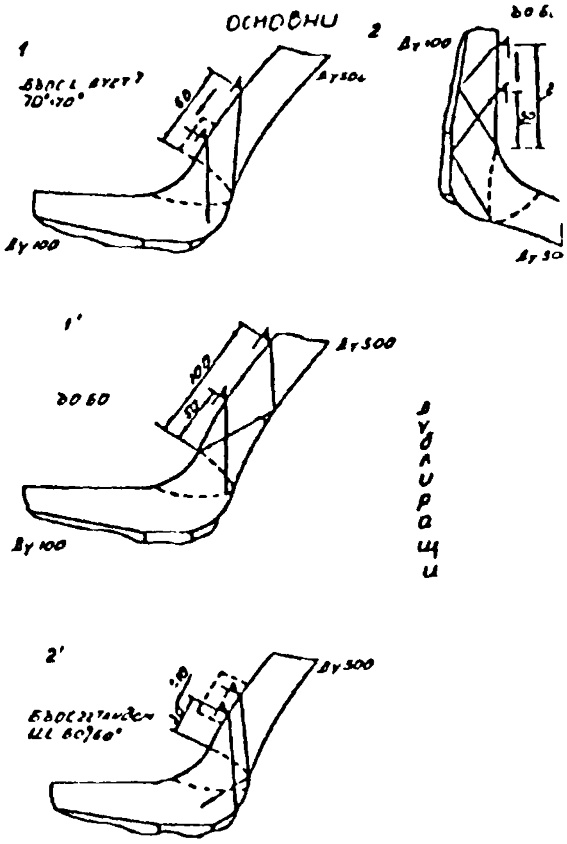
фиг. 6 а



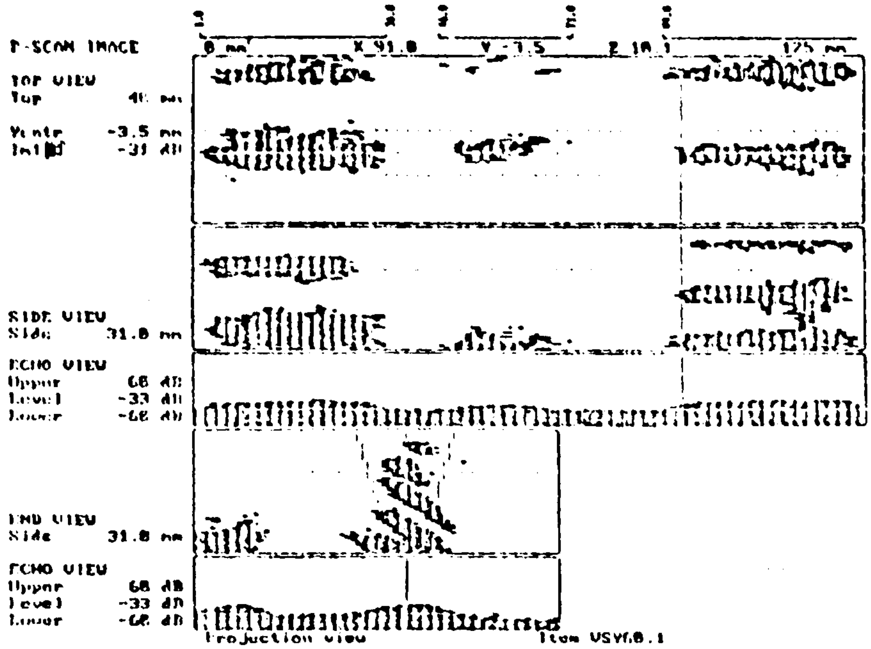
фиг. 6 б



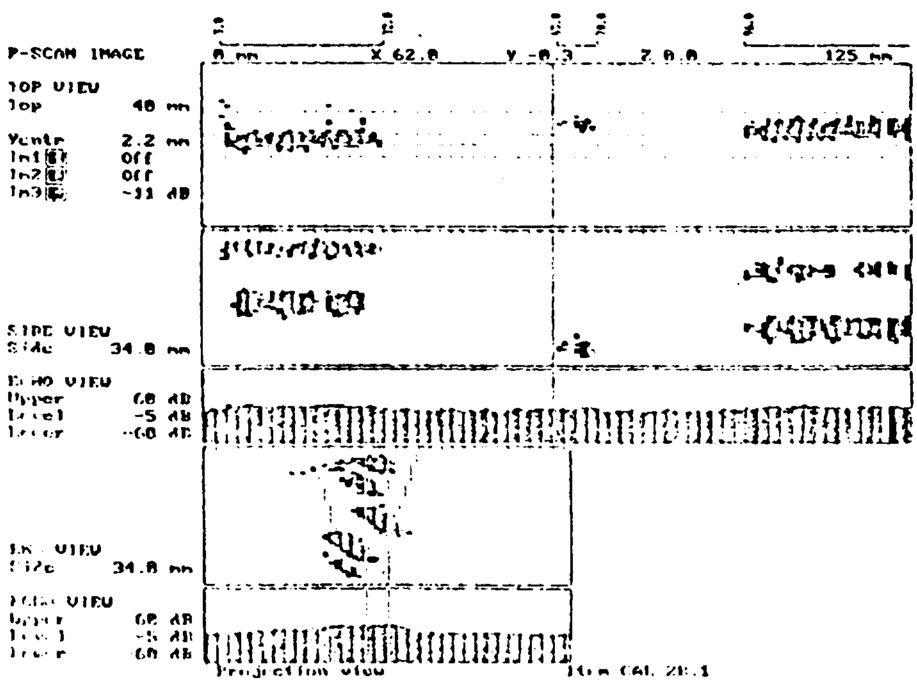
фиг. 6 в



фиг. 6 г


P-SCAN IMAGE
TOP VIEW
SIDE VIEW
ECHO VIEW
Upper 68 dB
Level -33 dB
Lower -68 dB
END VIEW
Projection view

фиг. 7


P-SCAN IMAGE
0 mm
X 62.0
125 mm
TOP VIEW
Top 40 mm
Ycntr 2.2 mm
SIDE VIEW
Side 34.8 mm
Projection view

фиг. 8

BG9700036

ВЛИЯНИЕ БОМБАРДИРОВКИ С УСКОРЕННЫМИ ДО 70 keV ЙОНАМИ НА СТРУКТУРУ И СВОЙСТВ АЛЮМИНИЕВОГО СПЛАВА

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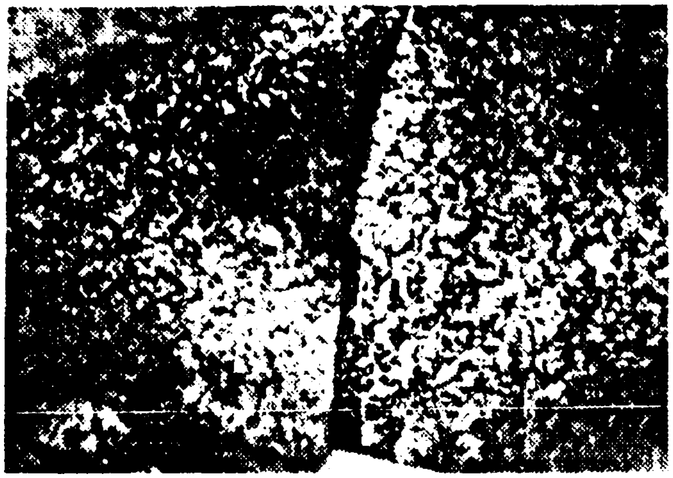
Алюминиевые сплавы считаются, рядом с нержавеющими сталями, семими подходящими материалами для первой стенки термоядерного реактора [1], [2].

Фиг. 1. Микроструктура (ТЭМ) быстрозакаленной ленты состава Al -1 вес. % Zn
а - 20000х; б - 40000х

Мы остановились на сплава Al-1 вес. % Zn как модель хомогенного твердого раствора на алюминиевой основы, из которого мы изготовили быстрозакаленных из расплава тонких лент толщиной около 50 мкм. Было исследовано влияние бомбардировки Fe^{+}- йонами на структуру и свойств лент. Йоны были ускорены до энергии 60 keV при дозах 10^{16} и 10^{17} йон/см^2.

В результат имплантации тяжелых железных йонов наблюдается как эфект эрозии поверхности, так и образувание новых частиц (вероятнее Al_3Fe) - фиг. 2 и 3.

Фиг. 2. Микроструктура (ТЭМ) быстрозакаленной ленты состава Al-1 вес. % Zn след имплантации Fe^{+}- йонами; 60 keV, доза облучения 10^{16} йон/см^2
а - 10000х; б - 100000х



Фиг. 3. Микроструктура (ТЭМ) быстрозакаленной ленты состава Al-1 вес. % Zn след имплантации Fe^{+} - йонами; 60 keV, доза облучения 10^{17} йон/$см^{2}$
а - 40000х; б - 100000х

Высокоэнергийное воздействие приводит к измельчению структуры - образувание субзерна около границу кристаллитов (фиг. 4).

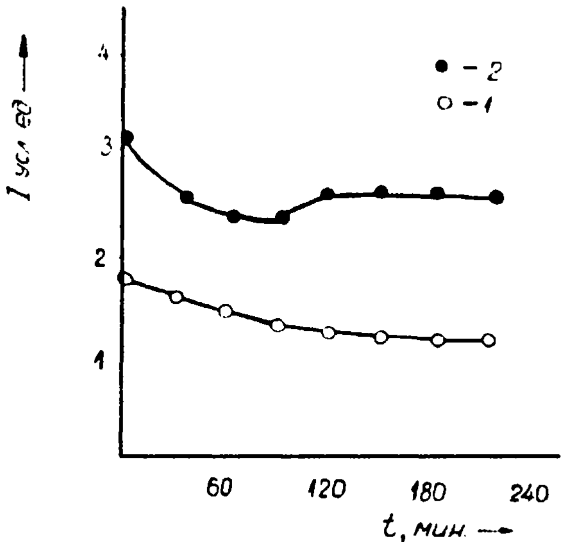
Фиг. 4. Микроструктура (ТЭМ) быстрозакаленной ленты состава Al-1 вес. % Zn след имплантации Fe^{+} - йонами; 60 keV,
доза облучения 10^{16} йон/$см^{2}$; 40000х

Было исследовано изменение магнитных свойств быстрозакаленных лент как результат имплантации железных йонов. Были проведены два типа экспериментов: 1) наблюдение хода намагничености I при изотермичном отжиге при 250 °C и 2) изохронный отжиг в температурном интервале 30 °C - 600 °C предварительно отожженных при 250 °C лент.

От вида кривых на фиг. 5 и 6 могут быть сделаны следующие выводы:

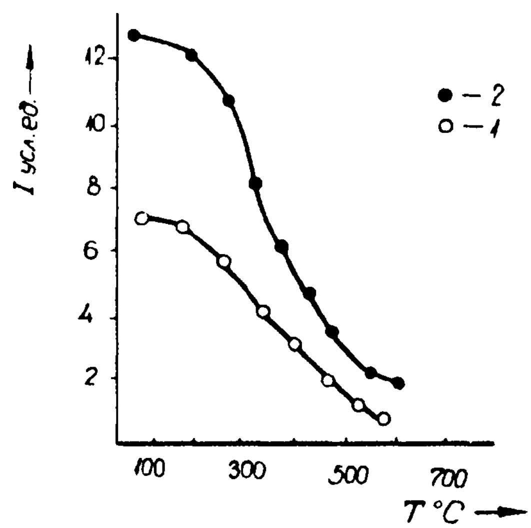
1) Существует надравновесная расстворимость примесей железа в быстрозакаленной ленте состава Al- 1 вес. % Zn;

2) Идет диффузия железных йонов в имплантированных лент с поверхности к внутренности преимуществено по границам зерен (фиг. 5, кр. 2), которая водит к повышению намагничености после 2 и больше часов отжига при 250 °C;



Фиг. 5. Изменение намагниченности I быстрозакаленной ленты состава Al-1 вес. % Zn в процессе отжига при 250 °С:
1 - лента после получения
2 - лента после имплантации Fe^{+}- йонами; доза

3) Поява квазиферомагнитного состояния имплантированного сплава [3], типичное для сильно разбавленных твердых расстворов, содержаЮщих феромагнитных атомов, с т. Кюри около 540 °С (фиг. 6, кр. 2).



Фиг. 6. Изменение намагниченности I быстрозакаленной ленты состава Al-1 вес. % Zn в процессе изохронного отжига скоростьЮ 17,3°С/мин:
1 - лента после получения
2 - лента после имплантации Fe^{+}- йонами; доза облучения 10^{17} йон/см^2, 60 keV

ЗАКЛЮЧЕНИЕ

Сплав Al-1 вес. % Zn является подходящим для проведения модельных исследований радиационных повреждениях алюминиевых сплавов.

Имплантация йонным ускорителем ИЛУ-4 является методом получения сильно разбавленных твердых расстворов с феромагнитными атомами, которые не могут быть получены обычними способами сплавления.

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BG9700037

Comprehensive Thermal-Hydraulic and Thermal-Mechanical Analysis of the Reactor Core and Fuel Rods for the Safety Validation of Real Refueling of VVER-440 at Kozloduy

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Introduction

The collaboration of the Section for Safety Analyses of Nuclear Power Systems and Kozloduy NPP started several years ago. The reason for it is the call for safety evaluation and validation of Units 1 to 4. Its aim is pursued by determination of thermal-hydraulic as well as thermal-mechanical margins of core and fuel rods. On the ground of numerical analyses performed by sophisticated computer codes the margins are determined for all required by the regulatory committee steady-state regimes and operational transients. The goal is to prove the existence of enough margins of the technological parameters to their permitted values.

The research results have to prove the safety and to show the optimal operation of Kozloduy NPP units during the choice of real core refueling. This aim is pursued by mean of existing word experience and implementation of computer codes for modeling, computation and analyses of thermal-hydraulic and thermal-mechanical processes.

Presented hereafter results are obtained under the „PROGRAMME 93-96 FOR UPGRADING OF THE OPERATIONAL RELIABILITY AND SAFETY OF UNITS I-IV VVER-440 (V-230) REACTORS". For 1994 the program foresees next three divisions have to be done:

I. Thermal-hydraulic calculations for the part of core with maximal heat flux for units 1 to 4 during steady-state regimes and the main circulation pumps coast-down during the earthquake situation. This part includes the determination of maximal fuel and cladding temperatures and critical heat flux ratio by subchannel code COBSOFM. The analyses are based on neutron-physical calculations and heat flux distribution in the fuel assemblies and rods.

II. Thermal-mechanical calculations and evaluation of the fuel rods behavior at core refueling of units 3 and 4 during steady-state regimes. The works include determination of temperature fields, fission gas accumulation under the cladding, changes in fuel rod's cladding etc. by computer code PIN-micro. The main input data for such analyses are the power history obtained from neutron-physical calculations and heat flux distribution in the fuel assemblies and rods .

III. Additional activities: 1) Preparation of manuals and implementation of computer codes COBSOFM and PIN-micro at Kozloduy NPP; 2) Training of plant employees to use the computer codes and performing of joint calculations and analyses; 3) Study of literature and consultations on processes and most optimal regimes of operation of fuel on the base of thermal-mechanical parameters; preparation of methodical recommendation to avoid fuel damage etc.

Computation of the core thermal-hydraulic processes by computer code COBSOFM

The code COBSOFM [1] is the modified version of developed in the US code COBRA-3C [2].
The main new features of this code can be summarized as follows:

* an improved model for the subcooled boiling with variable phases slip;
* pre-CHF and post-CHF heat transfer coefficients;
* a model for turbulent crossflow mixing in the two-phase flow;
* CHF correlation proposed by Russian authors for VVER fuel assemblies, that account the axial heat flux distribution .

The code permits to calculate the maximal fuel and cladding temperatures, the temperature and void fraction of coolant during steady-state regimes and operational transients. Critical heat flux ratio can be determined by Bezrukov-1976 [3] as well as by Smolin-1978 [4] correlation.

The code verification was carried out by comparison with experimental data obtained at Skoda plant (Check), Kurchatov,s institute and OKB Hydropress (Russia) experimental facilities.

Figures 1-6 present some of the results for 12th fuel cycle of unit 4. The most severe operational transient blackout of 6 from 6 main circulation pump under seismic circumstances is analyzed for the beginning, middle and the end of fuel cycle. The presented results include the coolant temperature, enthalpy, density and void fraction as well as core margins such as fuel and cladding temperatures and critical flux ratio.

On the ground of performed by COBSOFM analyses next conclusions can be made:

I. For steady-state conditions: the maximum of coolant temperature is 305°C at core outlet for the beginning of fuel cycle; there is no coolant boiling in the core; the maximal cladding temperature is 317°C that is under the design permitted for VVER-440 V-230 value of 350°C; the maximal fuel temperature is 1374°C that is also under the design permitted for VVER-440 V-230 value of 2808°C - the fuel melting temperature; minimal critical heat flux ratio is 2.197 for the beginning of fuel cycle, the hot channel factor 1.15 is already included in this value.

II. For accident with unit blackout: the maximal cladding temperature is 340°C that is also bellow the design permitted for VVER-440 V-230 value of 350°C; the maximal fuel temperature decreases from 1374°C for the beginning of the transient to 727°C 10 seconds after that is also under the fuel melting temperature; minimal critical heat flux ratio including hot channel factor 1.15 is 1.676 for the beginning of fuel cycle and it is higher than accepted value of 1.3. The last value ensures with 95% probability that there is no boiling crisis anywhere in the core.

The results presented here show enough safety limits of core as well as at steady-state regimes and during severe accident with unit blackout for the beginning, middle and end of fuel cycle.

Thermal-mechanical calculation of fuel processes by PIN-micro code

Since 1993 a team in the INRNE has began the application of code PIN-micro for the purposes of thermal-mechanical analyses of fuel rods during steady-state regimes and slow transients in the operating VVER-440 units in the Kozloduy NPP. The code is created on the base of the US code GAPCON-THERMAL-2 and it has been verified for VVER fuel.

Some specific physical models for the processes in the irradiated fuel rods such as densification, swelling and restructuring of fuel and cladding and irradiation creep of cladding are included.

By the code PIN-micro thermal and thermal-mechanical parameters of fuel rods can be calculated as follows: the temperature fields in the fuel, gas gap and cladding, gap heat transfer coefficients, fission gas release, gas pressure, mechanical stresses and strains in the fuel pellets and cladding etc.

The code has been verified against experimental data obtained at Kurchatov's institute research reactor MR and under the international collaboration program D-COM and FUMEX.

For the very first time in our country the code has been used for the calculations and analyses of thermal-mechanical parameters of VVER fuel rods. Figures 7-12 present the obtained results for the most loaded fuel rod in a working assembly. This assembly with typical power history that spent three fuel cycles in the VVER-440 core and is proposed for one cycle more to stay in the core. Average values of geometrical and technological parameters have been considered for this case.

The linear power of the assembly reached the maximum of 25 kW/m during the second fuel cycle, the same variable has the value of 20 kW/m during the third cycle. For the period under consideration the maximal fuel temperature remained below the value of 1000°C. The fission gas release is less than 0.4%, and the gas pressure under the cladding is 25 bars at the end of this period. The fuel and cladding radii have changed and this lead to the gap closure and the thermal contact between fuel and cladding after the middle of the third cycle. Obtained results showed that the fuel assembly with typical power history has enough safety margins (fuel temperature, gas pressure under the cladding) which ensure a safe operation during the considered three fuel cycles and the proposed fourth cycle.

Conclusion

On the base of presented results it can be concluded that codes COBSOFM and PIN-micro describe correctly the processes in the core and fuel of VVER and they can be recommended for safety analyses of this type reactors.

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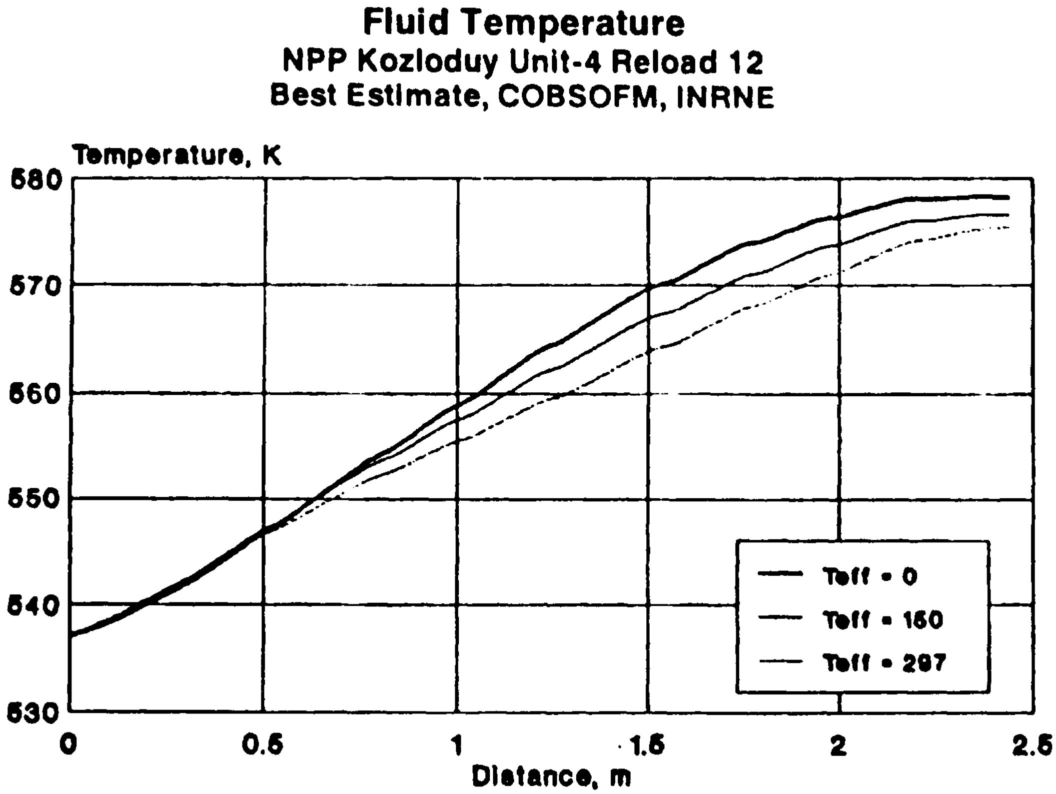
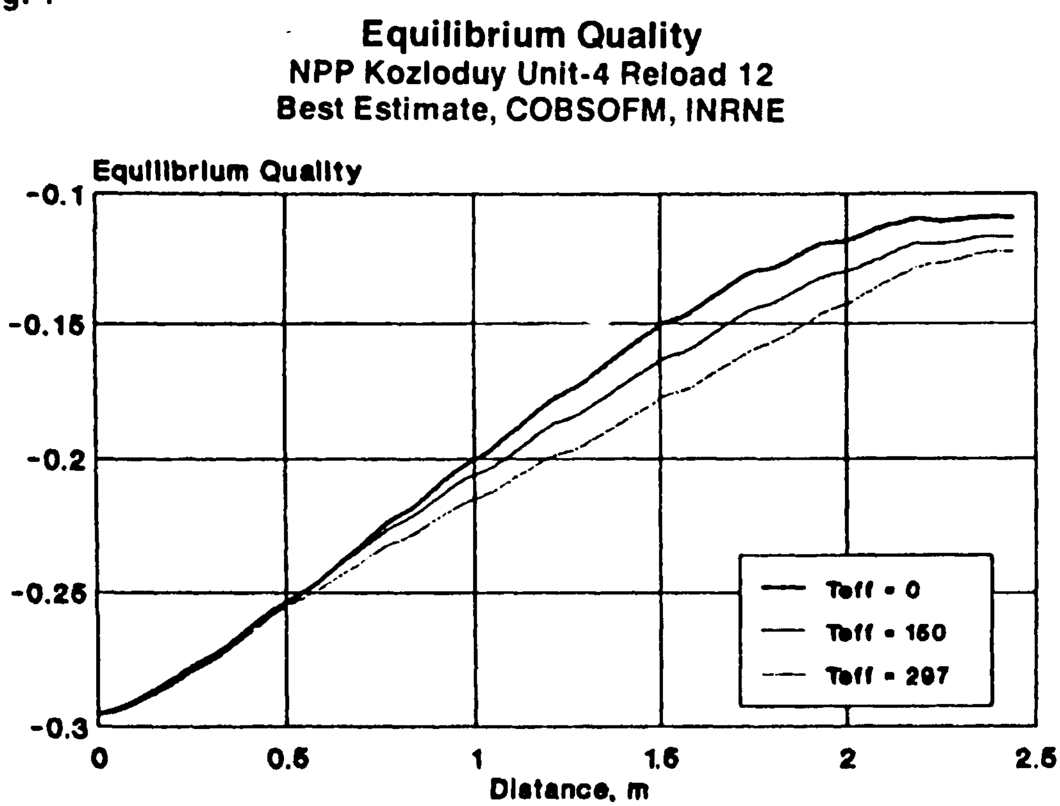


Fluid Temperature
NPP Kozloduy Unit-4 Reload 12
Best Estimate, COBSOFM, INRNE
Temperature, K
580
570
560
550
540
530
0
0.5
1
1.5
2
2.5
Distance, m
Teff = 0
Teff = 150
Teff = 297

Fig. 1



Equilibrium Quality
NPP Kozloduy Unit-4 Reload 12
Best Estimate, COBSOFM, INRNE
Equilibrium Quality
-0.1
-0.15
-0.2
-0.25
-0.3
0
0.5
1
1.5
2
2.5
Distance, m
Teff = 0
Teff = 150
Teff = 297

Fig. 2

Maximum Fuel Temperature
NPP Kozloduy Unit-4 Reload-12
Best Estimate, COBSOFM, INRNE

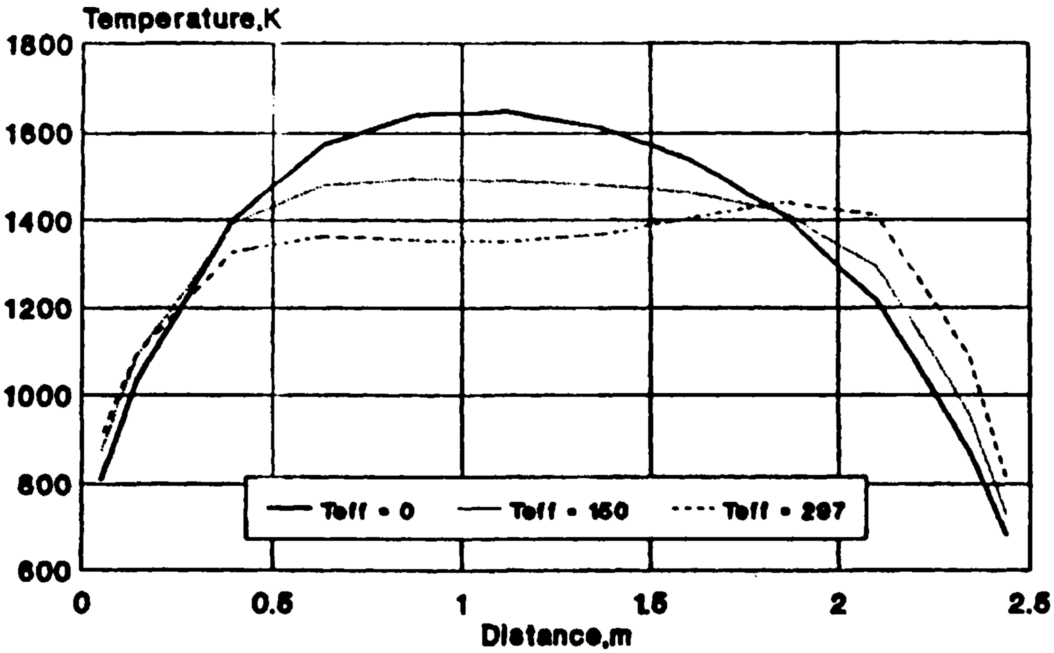


Fig. 3

Cladding Temperature
NPP Kozloduy Unit-4 Reload 12
Best Estimate, COBSOFM, INRNE

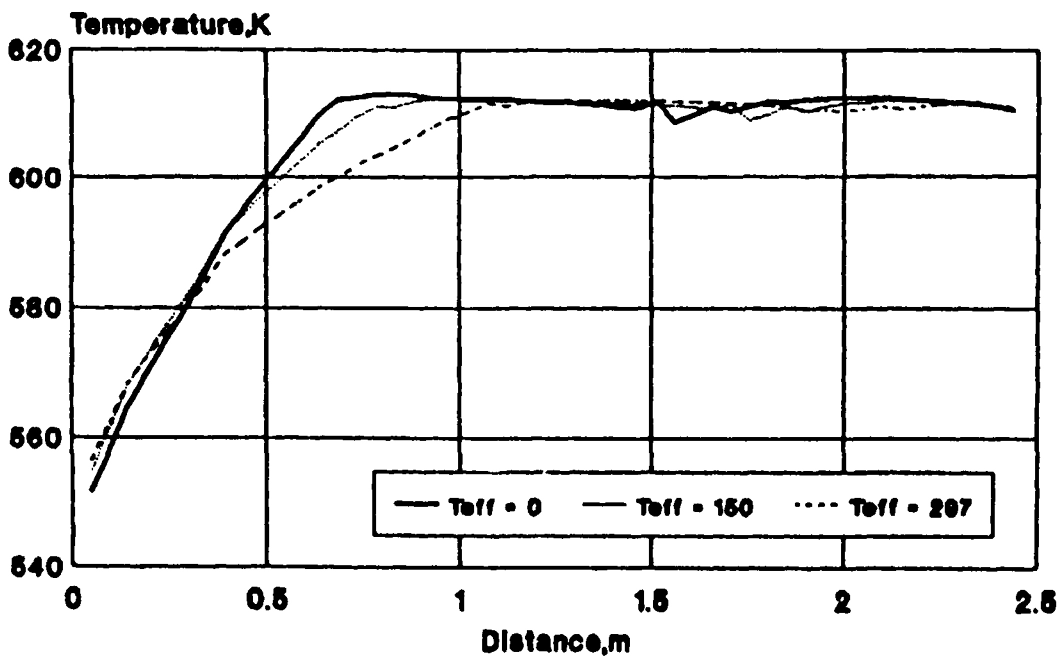


Fig. 4

Critical Heat Flux Ratio
NPP Kozloduy Unit-4 Reload 12
Best Estimate, COBSOFM, INRNE

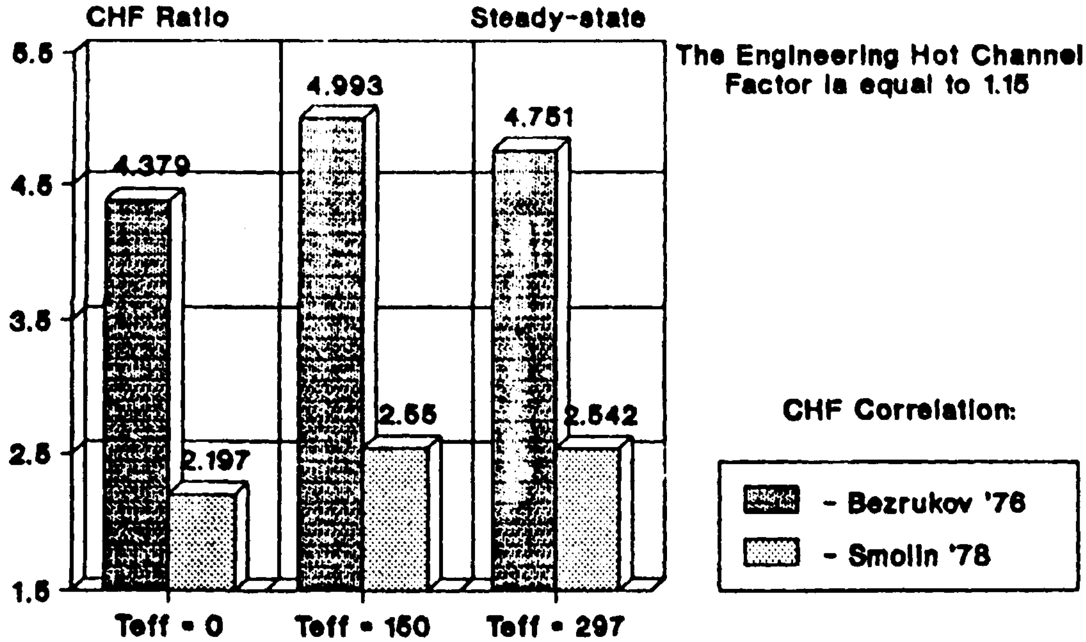


Fig. 5

Critical Heat Flux Ratio
NPP Kozloduy Unit-4 Reload 12
Best Estimate, COBSOFM, INRNE

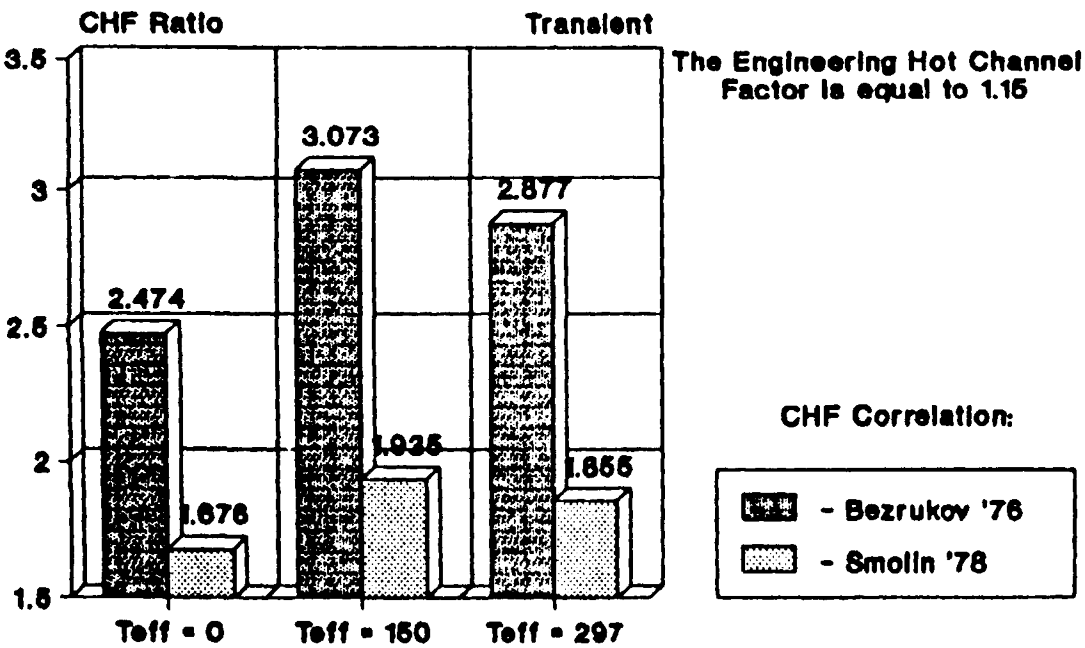


Fig. 6

Power History
WWER-440, Assembly 1

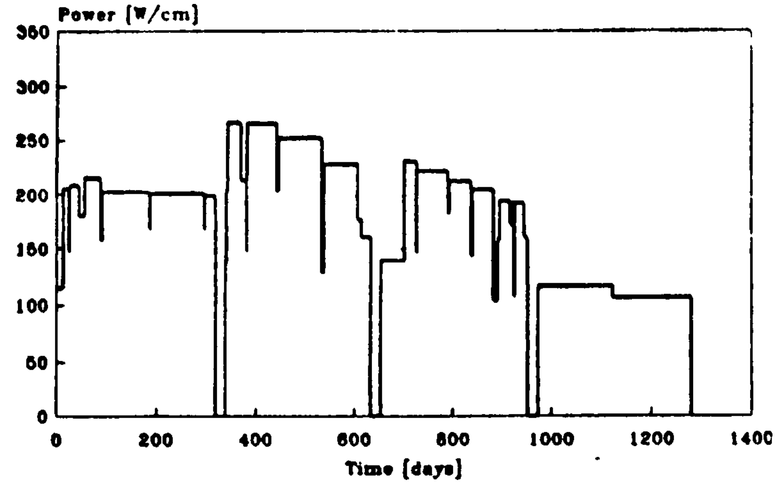


Fig. 7

Fuel Central Temperature
WWER-440, Assembly 1

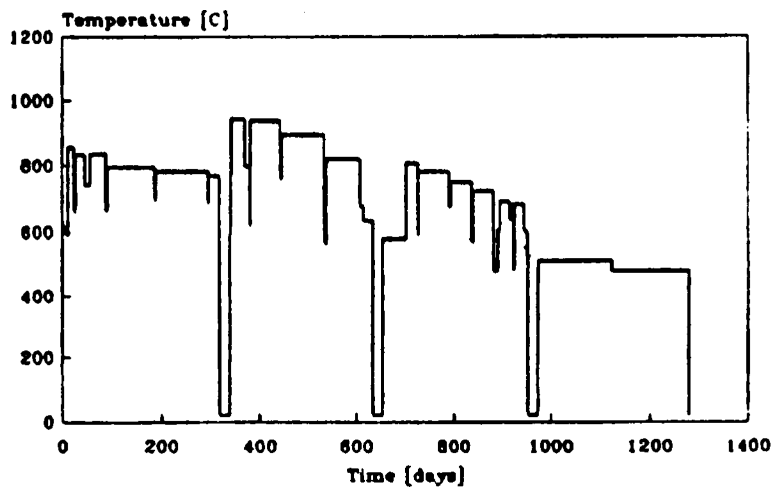


Fig. 8

Fission Gas Release
WWER-440, Assembly 1

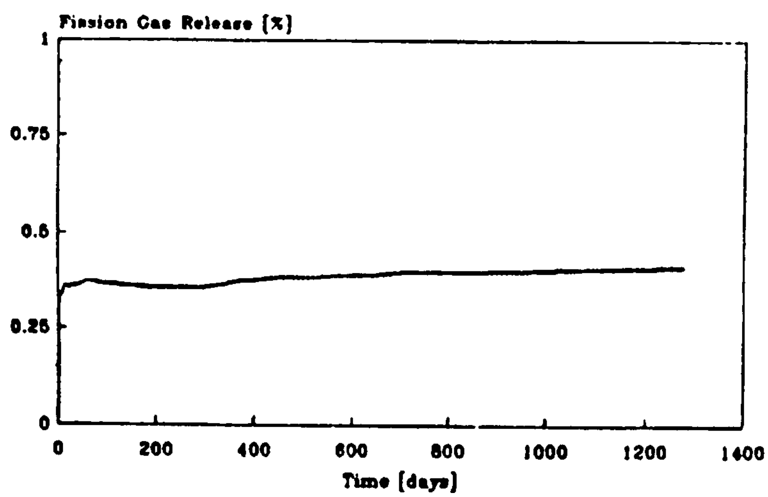


Fig.9

Gas Pressure
WWER-440, Assembly 1

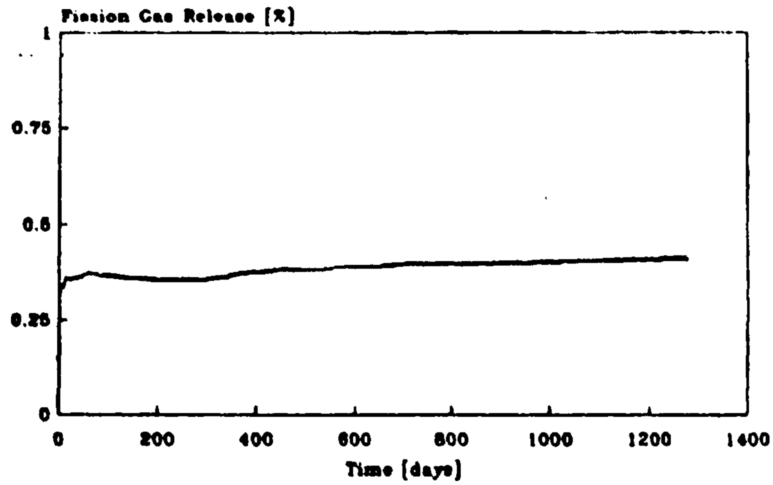


Fig.10

Fuel & Cladding Radii
WWER-440, Assembly 1

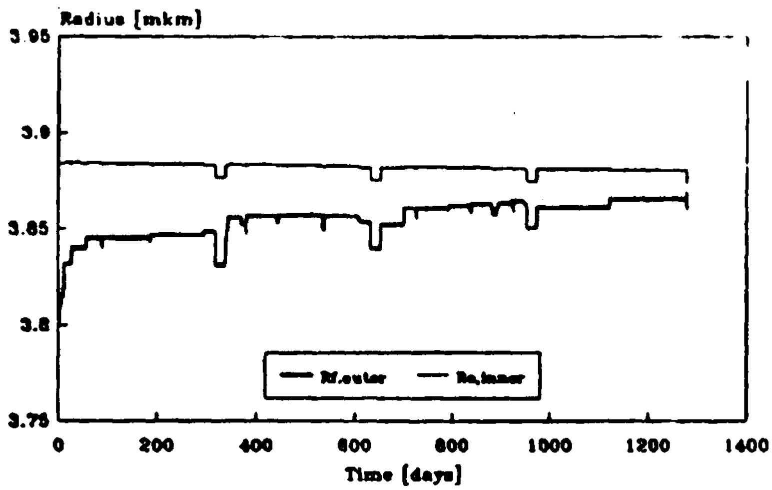


Fig. 11

Fuel Central Temperatures
WWER-440, Assembly 1

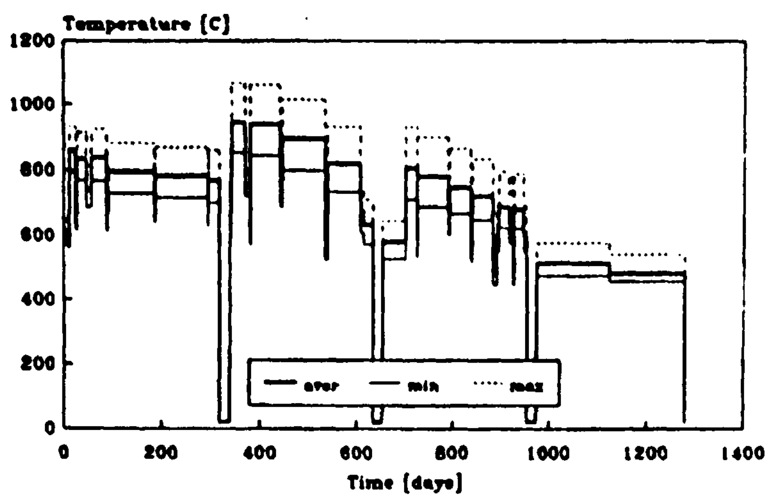


Fig. 12

BG9700038

RADIOACTIVE WASTE PROBLEMS IN KOZLODUY NPP

N. Videnov, V. Stanchev,

KOZLODUY NPP

The radioactive wastes /RAW/, liquid and solid, are product of the daily operation and are associated to the production of electricity from NPP all over the world.

The annual quantity is a function of the number and the type of units, outage programme, emergency maintenance, reconstruction of the units and the culture of operation of the personnel working in the restricted area.

By design for WWER - 440 units it is foreseen storage of liquid and solid wastes only in the specified for this purpose temporary storage facilities and periodically these facilities should be enlarged in order to take the new generated quantities of wastes. There are no such enlargements to date and the efforts of the specialists and KNPP management to solve the problem of increasing the radwaste volume with temporary technical decisions are obvious.

By design the WWER - 1000 should be supplied with bitumen installation for treatment and packing of liquid RAW, system for incinerating the burnable solid RAW and compaction of the other wastes - about 50 tons force. Besides, the bitumen storage facilities are planned for not more than 5 year units operation. For the incinerated and compacted solid wastes storage facilities are not foreseen. Generally, these design decisions did not solve the major part of the radioactive waste problems of WWER - 1000 units. In addition, the licensing of the bitumen installation and the RAW incineration system in front of the regulatory bodies in the former USSR failed, the delivery of the installations was canceled and thus there was no reason for project fulfillment.

The difference of the WWER-440 and 1000 and the power units used in West Europe NPPs is the generation of great quantities liquid RAW from WWER operation in return to the very low volumes of spent resins which determined the choice of treatment technologies, packing and temporary storage on site.

The management of NPP Kozloduy evaluating the significance of the operation with the RAW started establishing of structures for RAW managing in 1991 and since April 1992 the Department for Treatment of Radioactive Wastes appeared. The Department is a part from the NPP Kozloduy structure. It runs all buildings and facilities which treat and store the RAW and also the spent fuel storage facility.

The main principles which are followed in the activities related to the RAW are:

- providing, at maximum, nuclear and radiological protection of personnel, population and environment.
- optimum economy (efficiency) in managing the RAW.
- implementation of modern technologies for treatment and temporary storage of processed RAW and decommissioning of nuclear facilities which technologies has proved their efficiency in the practice.

At this stage the RAW management includes activities associated with solid wastes such as collection, preliminary sorting on radiological and physical characteristics, transportation, precompaction in drums and supercompaction of these drums. The activities associated with the liquid RAW and the contaminated oils will be developed in the near future and are described further down.

I. PROCESSING OF SOLID RAW

After reaching design power in Kozloduy NPP annually is generated average volume of 1400 m^3 solid RAW but the biggest volume has been produced in 1993 - 3100 m^3. The reason for this is the sequence of the V and VI unit commissioning and the reconstruction performed mainly on I and II power unit. The tendency for 1994 is decreasing the production of wastes which is expected to reach about 2000 m^3. This process will continue in the future because of the developed programmes for minimizing the generated solid wastes.

The generated solid RAW from the KNPP units are transferred to the permanent and temporary places in the Restricted Area, in which are placed wastes with gamma activity not higher than 0.30 mSv/h. For RAW with activity from 0.30 mSv/h to 2.00 mSv/h and from 2.00 mSv/h to 9.00 mSv/h are manufactured and tested special containers with the necessary biological protection which provides safe personnel operation during container filling and transportation. The location of this places depends from the unit condition - normal operation, outage or emergency maintenance or reconstruction and this location is coordinated between the Electricity Production Depts. and the "Treatment of RAW" Dept. These places are used for preliminary sorting by radiological and physical characteristics. For transportation of the collected RAW is used especially equipped truck which brings the wastes to the processing place. Additional sorting is carried out, the aim of which is optimization of the ratio between the so called soft RAW (contaminated clothes, plastic packages and other materials used during the repair activity) and the solid - metal, wooden parts and other in the drums. Well known are the problems which are caused by the soft wastes during precompaction and supercompaction. If the drum is full with soft wastes only after supercompaction this drum returns to its previous volume and there are cases in which the upper rolled lid opens which causes difficulties in the technology process and increases the final volume for storage.

After additional sorting the wastes are placed in 200 l. steel drums and are precompacted with a compaction force of 15 tons. In this way is obtained decreasing of the volume in advance by the factor of 3 to 5. Usually a 200 l. drum can take 0.8 m^3 RAW. After rolling the upper lid the drum is then taken for final compression in a compactor with greater force. Since December 1991 till February 1994 the drums have been compacted with a compaction force of 630 tons. Since February 1994 after successful site acceptance tests the Supercompactor, delivered on contract with Westinghouse manufactured by BLISS Company, was put in operation. The Supercompactor operates with a force of 1000 tons and provides final wastes volume decreasing, including precompaction by the factor of 7 to 15. For the Supercompactor is provided a temporary facility in the Restricted Area in Auxiliary Building -3 and is additionally equipped with feeding and discharge drum line designed and manufactured in "Treatment of RAW" Dept.

II. PROCESSED RAW STORAGE

After Supercompaction the RAW are placed in special baskets and in this condition transported for temporary storage in the processed RAW storage facility. This temporary storage facility is located in the Northeast part of the NPP, in the physical fence, far enough from the other facilities and meets all the IAEA recommendations and the national legal standard requirements. The temporary storage facility is licensed by the ISUAE (Institute of Safety Usage of Atomic Energy) of CPPUAE. In addition around the storage facility there is an alarm system and the access is limited only for the "Treatment of RAW" and the Control Bodies personnel.

III. IMPLEMENTATION OF THE PROJECT FOR TREATMENT AND TEMPORARY STORAGE OF PACKED LIQUID AND SOLID RAW

There are four main technologies for waste treatment and packing involved in the project.

1. Technology process for treatment of solid RAW which comprises the following systems and installations:

- Waste retrieval from the storage near the switch gear yard

For this activity the Technical Project foresees construction of movable facility equipped with all necessary technical means for waste retrieval from each storage pit thus it is provided to maximum extend process mechanization. Manual operation will be carried out only in exceptional cases during the preliminary waste sorting. The facility meets all the personnel radiation protection requirements and provides the necessary labour conditions. The facility project, construction and equipment manufacturing is being performed by the Shipbuilding Institute - Varna.

After removal of all solid RAW from the storage in the future it will be used as a buffer one and a reserve storage volume during units decommissioning.

- Waste retrieval from the Auxiliary Buildings (AB) - 1, 2, and 3 storages

The work here will be performed on three stages. The first stage includes construction of superstructure of the open area of AB-2 and waste retrieval from this superstructure, the second stage includes the same activities for AB-1. The equipment supplied by Westinghouse will be used for retrieval of the wastes and the design activities are Energoproject part. The third stage is waste retrieval from AB-3 after building of new or adjustment of the present facilities because of the storage specialty in AB-3.

All activities are remote controlled except the preliminary waste sorting.

- Waste transportation to the treatment facility will be performed by means of container trucks "LIAZ-MADARA" equipped with a device for self loading and unloading of the container in the storage and the treatment facility.
- the waste processing in the treatment facility will be a separate technology process including several subsections:
  - section for breaking-up the oversized wastes coming from the storages
  - section for decontamination of metal RAW
  - section for storage of wastes which radiation activity is > 2 mSv/h and afterwards they are directly transferred for packing.

The main technology process includes several associated operations for 50-65 drums per shift output such as final sorting by radiological and physical characteristics, precompaction - 50 tons compactor, rolling the upper lid of the drum, weight measurement, radio nuclide content and total activity of each drum and transferring the data in an information control system, intermediate station where sorting of the drums by activity is performed, 1000 t. Supercompactor for drums compaction, second station for pucks (compacted drums) and a steel - concrete container BB-cub.

All operations in the main technology process are mechanized by means of remote control and there is a possibility for automization of these operations, except the sorting area. Generally said, there is no way of mechanizing the sorting operations and we are not informed for any sorting systems applied in the world practice. If there are such systems probably they are used by highly developed and rich industrial countries.

2. Technology process for treatment of liquid RAW

It includes retrieval of the liquid phase and dilution of the crystal phase of the liquid wastes from AB-1 and 2 and their transferring in a tanker-trailer which transports the wastes to the treatment facility. The transferring of the liquid waste from AB-3 is carried out by pipe system because of their immediate proximity.

The treatment of the liquid waste is performed by Westinghouse formula, technology and solidification system. The main advantage of the chosen technology is the possibility of concentration by evaporation of the liquid waste from base position of 13-15 % salts to 40 % in the evaporator outlet concentrate. In this way we reach final cemented product decreasing by 0.74 coeficiency while the coeficiency of the volume of the final cemented product in the normal solidification technologies is increased by the factor of 2-2.5. As a final result the implementation of this formula and technology leads to significant decreasing the number of packages and the volume for temporary storage and final disposal of the treated RAW.

The solidification system is completely automatic.

3. Technology process for packing of the treated solid and liquid RAW

This process includes containers' transportation to the separate, the so called stations, at which the following operations are performed: loading the container with pucks, filling the free space in the container with cemented waste (here is possible a second option which allows if necessary filling the container only with cemented waste), placing and welding the container lid, 24 hours stay for preliminary solidification of the mixture, container air tightening and transportation out of the treatment facility to the temporary storage facility. All operations are remotely controlled and there is a possibility of process automization.

The container which is used in the project is jointly developed between Kozloduy NPP and BalBok Company. It is licensed by CPPUAE as type package IP-3 which gives the opportunity in future to be transported on the national road net to the place specified for final RAW disposal.

4. Oil incineration system

The Oil incinerator burns the contaminated oils produced during Kozloduy NPP units operation. The system is mobile, it is not in the frames of the treatment facility and can be installed on each unit according to the necessity. It is completely automatic.

The storage of the treated and packed RAW will be done in temporary storage facility which is located next to the treatment facility. The unloading and stacking of the containers will be carried out by cranes, specialized clamps and co-ordinate system, remotely controlled.

The Technical design of the RAW treatment facility which main designer is Energoproject, is approved by the Expert Council in Kozloduy NPP and is transmitted to the regulatory bodies, Committee for Peaceful Purpose Usage of the Atomic Energy, Ministry of Environment, Health Ministry, Ministry of Internal Affairs (HFP) for review, statements and the associated construction permissions. The Health Ministry has given its agreement, the Ministry of Environment conducted an open discussion on the project in Kozloduy municipality and in the Ministry in Sofia, we expect the review by Expert Council in the Ministry of Environment, also we expect the statement of CPPUAE after reviewing the independent project expertise by the Bulgarian Academy of Science (BAS).

IV. RAW MANAGEMENT IN KOZLODUY NPP

All the activities above stated, performed in Kozloduy NPP, are part of the developed and fulfilled on stages two years long "Programme for RAW Management". This programme foresees development of several number of internal regulation documents which will include all not covered to-date stages of RAW management including minimizing the generated quantities solid and liquid RAW, the interaction between the Electricity Producing Departments, the outside organizations and the "Treatment of RAW" Dept., feasibility study and new technology development for RAW treatment.

The lack of modern national regulations creates to a great extend difficulties in the RAW management.The documents which are in force now are Order No. 0-35 of the HM and MIA - 1974, Order No 46 of the HM and CPPUAE - 1976, Order No 7 of CPPUAE - 1992 and ONRZ-92 of CPPUAE - 1992. There are serious contradictions on one and the same item in the texts of the regulations which gives the opportunity of different interpretation from each regulatory body and the users. It should be noted that there is no complete regulatory document dealing with the RAW management from the moment of their generation to their final disposition containing concrete actions, authorizations and responsibilities of all organizations involved in this process. Probably after Standard's No 111-S-1 "Developing of National System for RAW Management" approval by the IAEA Managers Board, according to Articles 12 and 13 of the Law for Peaceful Usage of Atomic Energy (LPUAE), under the supervision of CPPUAE it should be organized a commission which will prepare the list of the necessary regulatory documents, to be discussed by the involved organizations and the developing of the regulatory documents to be assigned to competent specialists.

According to us, it is obligatory funds to be established for RAW management on three levels - Kozloduy NPP, National Electric Company (NEC) and national one. There are established such funds in several European countries as for this purpose are allocated 5-10 % from the electricity cost price produced by the NPP.

Immediately after approval of the new regulatory documents should start practical actions for their fulfillment. The most important document, in our opinion, is the one treating the issue with the final disposal repository for low and intermediate RAW.

The practice in the countries which already run such repositories (France, USA, Germany, Great Britain, Spain and Finland) shows that from the moment of creation of the necessary document till its practical realization pass about 10 - 15 years. Bulgaria should hardly be an exception. If we suppose that the criteria and the requirements for repository for final disposal of RAW will be available in 1995 (an optimistic prognosis) we will have repository not earlier than 2007. One of the most serious problems on RAW management in Kozloduy NPP comes from here. The storage facility that is to be constructed can take about 2000 containers which means 2-3 years treatment facility operation. Besides, it will be treated and stored only two thirds ($^{2}/_{3}$) from now existing liquid and solid RAW. For the rest part of the wastes and these that will be generated from now on there is no storage room on site.

The only way out, that we think is really possible, is design and construction of long term storage facility for treated and packed RAW which to be located on the alienated areas Eastwards or Northwards from the NPP. Our intentions are in the frames of the two years long "Programme for RAW Management" to assign feasibility study for siting the area and the way of waste storage. The results from this study will be presented in front of the regulatory bodies for their statement and our intentions are to present these results also to the competent organizations for discussion.

BG9700039

STORAGE OF SPENT NUCLEAR FUEL: THE PROBLEM OF SPENT NUCLEAR FUEL IN BULGARIA

Z. Boyadjiev[1], E. I. Vapirev[2]

Introduction

The storage of spent nuclear fuel (SNF) is in intermediate stage of the nuclear fuel cycle - between the utilization of the fuel in a NPP and the reprocessing or final disposal.

The storage of SNF requires special technologies because of the high radioactivity (~1 Ci/g, ~10+5 n/s.kg), the residual heat (~1W/kg), the generated plutonium (5-10 kg/t U) and the hazard of radioactive contamination in case of fuel degradation.

The SNF is a special type of radioactive product generated during the work of nuclear power plants. The common feature between SNF and highly radioactive wastes (HRW) is that the planned final repositories are both for SNF after preconditioning procedures and HRW.

1. The problem of SNF.

Up to 1993 more than 120000 t heavy metal (tHM) of SNF from LWRs and HWRs have been generated. Only 5000 tHM have been reprocessed without the fuel for ~220 t military plutonium. [1,2] The expected capacity of the reprocessing plants in 1995 is ~ 5800 t/a at generation rate of ~10 000 12 000 tHM/a. If the new plant in Russia is put into operation the reprocessing capacities will increase with two lines of 1500 t/a each. The plant in Krasnoyarsk is planned for VVER-1000 fuel but its construction is temporarily stopped (1994).

The amount of generated fuel worldwide is greater than the total capacity of all reprocessing plants but nevertheless the plants are not loaded the reprocessing plant in Sellafield has commenced work at the beginning of 1994 although it has been in condition for operation for a long time. The reason for the postponement is that the contracts for reprocessing do not guarantee continuous and profitable process. The high reprocessing price makes most of the potential customers to defer the decision for reprocessing or disposal.

The estimated reprocessing capacity for LWR fuel is summarized in Table 1 [3] :

Table 1

SNF Reprocessing capabilities, tHM:

	1992	1995	2000	2005	2010
France	1200	1600	1600	1600	1600
UK	0	1200	1200	1200	1200
Russia	400	400	400	400	400
Japan	100	100	900	900	900
total:	1700	3300	4100	4100	4100

The capacity of the reprocessing plants after the year 2000 is expected to be able to handle the fuel from ~160 reactors of the VVER-1000 type.

Approximately 50% of the total quantity will be generated in countries which have accepted or are close to accepting the option for final disposal without reprocessing: Canada, Sweden, Spain, Finland, USA.

Another group of countries support the reprocessing option: UK, France, Belgium, Argentina, China, Italy, Russia, Germany, Japan etc.

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Japan and other countries have invested in research projects for reprocessing of SNF with partitioning of Pu, Am, Cm and other transuraniums, and also of long living fission isotopes (Cs-137, Sr-90). Both options are investigated for final disposal of the separated isotopes and also transmutation of the long living isotopes fissioning with fast and thermal neutrons for the actinides and photonuclear reactions for the Cs and Sr. Most of the efforts are for the plutonium isotopes [4,5].

The existing stockpiles of plutonium are planned to be reused in MOX fuel. In Japan, France, Belgium and other countries there are facilities for production of MOX fuel. The estimations are that the mixed oxide fuel can reduce the rate of plutonium generation by 1/3 and the already separated plutonium can be "burned" in the beginning of the next century. 30 reactors in Europe have already been licensed for use of MOX fuel.

A third group of countries without reprocessing or repository capabilities store the SNF in wet or in dry storage facilities and have delayed the decision waiting for further development of the technologies and reduction of the cost of the back end. Approximately 14 countries have accepted the policy of "deferred decision", 9 of them together with the option for reprocessing or final disposal. (Argentina, Czech republic Finland, India, Italy, S. Korea, Lituania, Mexico, Pakistan, Slovenia, S. Africa, Spain, Ukraine)
Bulgaria in an IAEA report is attributed to the group of countries with deferred decision [6].

The greater part of these countries have constructed or planned facilities for storage of SNF for all the fuel which is expected to be generated until the end of the operation period of the NPPs .

The conclusion from the present paragraph is that:

a) although the capacity of the reprocessing plants is more than two times less the generation rate of SNF, the plants are not loaded. Approximately only 5% of the total amount of generated SNF has been reprocessed;

b) the number of countries which have delayed the decision for the back end of the nuclear fuel is significant;

c) no big quantities of SNF have been diposed -off in final repositories. The first repository is expected to start operation near the end of the centrury.

OR: there is a demand for SNF storage capacity.

2. Technologies for SNF storage.

The existing technologies are wet and dry storage of SNF [7,8,9].
The wet storage is an already developed technology and there are no communications for fuel degradation during wet storage. For dry storage of uranium dioxide fuel inert atmosphere is necessary for temperature above 150 C since for failed fuel pins additional oxidation occurs and the cladding can be damaged.
In the last years new technologies were developed for dry storage of LWR fuel in inert atmosphere, double barrier, fuel control and passive cooling.

2.1. Wet storage of SNF [7,8].

The wet storage is only for intermediate storage of SNF. The projected terms of storage is in average 40 y, the maximal term is ~60 y.
The wet storage is considered to be an already developed technology. Some work is still continuing on consolidation of fuel, criticality calculations with Keff=0.95 and further nuclear safety improvements.
The quality of the water is maintained by:

- on exchange systems;
- filters;
- systems for cleaning the surface of the water;

- sucking systems for the bottom of the pools;
- systems for scrubbing the walls of the pools, especially at the boundary air/water.

In most of the wet facilities there is a space between the liner and the concrete for leak control;
The fuel consolidation with boron absorbers between the assemblies is a practice both for AR and AFR pools.
The radioactivity release in the environment from wet facilities (AFR) is negligible and in some cases it is beyond the limit of detection.
The major radioactive wastes from AFR pools are ion-exchange resins.

The basic disadvantages of the wet technology are:

a) active systems for supply and purification of water;

b) possibility of loss of water for beyond project accidents. In case of loss of water the fuel can be damaged from overheating.

2.2. Dry storage of uranium-dioxide SNF [9,10,11].

Dry storage with passive cooling with air has the inherent possibility for storage for more than a 100 y. The present technologies are for terms of 40-60 y. The declared terms of 40-60 y is due mainly to the lack of experience for long storage. There are communications for projects over 100 y [10].
At present the dry storage terms are comparable with those for wet storage . If the storage period is increased over 100 y that will be a strong advantage over the wet technology.
The developed technologies for dry storage make possible safe storage in regions with increased seismicity.
The major deficiency of the UO_2 fuel is the possible further oxidation to U_3O_8 when stored in air and high temperature. If there are defective fuel pins the cladding can be damaged since U_3O_8 has less density than UO_2 . When the fuel is further oxidized the original crystal lattice changes and the ability to retain the fission fragments decreases rapidly.

When the fuel is oxidized to U_3O_7 the volume of the fuel is not changed since the densities of UO_2 and U_3O_7 are the same. When the temperature is lower and the content of oxygen is less the fuel is oxidized to U_3O_7.
For temperature below 300C the rate of the reaction decreases by an order of magnitude each 30 degrees [12,13].

According to a 1990 research safe temperature for storage of defective fuel in air for 30 y is 135-160C [10]. For inert atmosphere with 1% oxygen for 30 y the upper limit of the safe temperature is estimated 210-220C. There are more general estimations that probably acceptable temperatures for which the oxidation of SNF will not lead to observable effects is below 200C [9].

According to 1992 estimations non-defective fuel can be safely stored in air for 30 y at 330C [11].

The possible solutions for dry storage of SNF are:

- storage in containers in inert atmosphere;
- storage of defective fuel in inert atmosphere;
- storage of the fuel below the oxidation threshold temperature.

The basic principles which have to be observed when facilities for dry storage are designed are:

- two independent barriers;
- possibility for fuel control;
- possibility for fuel retrievability.

The developed technologies for dry storage are divided into two major classes - double purpose technologies - for transport and storage, and technologies only for storage.

Typical representatives of the double purpose technology are the CASTOR and TN-AVR casks. The casks are filled with helium, have partial shielding for gamma-rays and neutrons. The seal is double with the option for control of possible helium leak between them. The stored in vaults casks are cooled passively by air. In the last years modular facilities with passive cooling for dry storage of LWR fuel with burnup up to 50 MWd/tU have been developed.
The modular facilities are divided into two types - vault types in which the increment is in large "portions" and facilities which consist of small modules for 1 - 4 canisters for 15-20 fuel assemblies each.

Modular vault facility is the storage facility built by GEC Alsthom for gas-cooled reactors. Similar facility is the one in Cadarache although not modular (the proposed by SGN facility is modular). In these facilities the fuel assemblies are sealed each in a separate tube in helium or nitrogen atmosphere and one or several such tubes in one common vertical larger sealed tube passively cooled by air. Such a system allows the control of possible helium leaks in the larger tube. GEC Alsthom Engineering Systems ltd have a license for modular vault dry storage facility (MVDS) in Fort Saint Vrain (USA) and also a proposed project for Paksh. The operations in the MVDS are remote and automatic.

A typical representative of small modular facility with passive cooling with air is the NUHOMS system. In USA by NUTECH (beginning 1980 r., NUTECH, DOE, Electric Power Research Institute, Carolina Power&Light) a horizontal modular system for SNF has been developed (Horizontal Modular Storage - NUHOMS). The system is passive and consists of two modules - heavy shielding module - HSM and dry storage canister - DSC. Because of the modular type and cheap materials (concrete for HSM) it is considered that the system has economical advantages compared to other technologies. The modular type makes possible the building of a high capacity facility without considerable initial investments.

The dry canister is a stainless steel cylinder in which there is a gasket with sleeves for the assemblies. The leak tight bottom is guaranteed by the in-factory checked welding. The front end is sealed by two independent welds performed by automatic welding machine. During the fuel storage period the condition of the fuel is not controlled. The ihermal capacity of the DSC is 20 kW or 24 fuel assemblies from VVER-1000 can be stored in the DSC after 5 y cooling in AR or AFR pools. The cooling gas is either helium or nitrogen. The approximate collective exposure is 10 mSv.man per cask transportation and loading in HSM.

There are communications (1993) for further development of the system in Japan for better seismic safety.

In 1993 a similar system was reported developed by AECL with vertical storage of canisters in concrete modules [11]. The MACSTOR system includes storage canister, transport cask with gamma- and neutron shielding and modular facility for vertical storage. The system MACSTOR is designed for LWR fuel which is a new field for Canada and therefore Nuclear Fuel Handling Services. Transnuclear Inc. (TNI) are consultants for MACSTOR.

In 1994 there were reports for a development of a multi-purpose cask, MPC, for transport, storage and final disposal [14]. The DoE of USA supports the projects mainly because of incompatibility of different systems for storage and transport which requires additional fuel reloading, inefficient use of equipment and generation of contaminated and activated equipment. The additional fuel operations increase the cost of the fuel back-end.

3. The problem of SNF in Bulgaria

At the signing of the contracts for the nuclear power reactors in Kozloduy the Soviet partners took the obligation for returning back of the generated SNF for the life of the plants. These obligations were strictly observed for nearly 10 y and all generated spent fuel assemblies 3086 were returned to the Soviet Union. In 1979 the Soviet side insisted on building of a AFR facility which was to be put in operation in 1985. The purpose of this intermediate storage facility was to give time for the construction of the reprocessing plant in Krasnoyarsk. In 1987 the Bulgarian application for SNF export was denied since there was no free storage pools and the construction of the reprocessing plant had not yet started.

The AFR facility in Kozloduy was put in operation with a 3 y delay and the Soviet Union accepted SNF as an exception in 1988.

At the moment (Nov.1994) approximately 1800 fuel assemblies are stored in the AFR facility and some 1060 in the AR pools.

The short description of the problem shows that at present Bulgaria does not formally belong to the group of countries with "deferred decision". Deferred decision means that the decision for reprocessing or final disposal will be taken after a certain period e.g. 30 -50 y. At present the national policy is to export from the county of the SNF (in our case for reprocessing). No decision has been taken for building of facilities for intermediate storage for all SNF which is expected to be generated till the end of the plant life.

There are certain economic advantages of the policy of "deferred decision" for a part of the SNF. A very careful analysis has to be performed for the various options of the "deferred decision", most probably with dry storage technologies, which in the end will include reprocessing or final disposal. The decision concerns approximately 12000 fuel assemblies for a term of 40-50 y.

When the technologies are estimated besides the nuclear safety features some additional parameters have to be considered:

- compatibility of the possible technologies for transport to the reprocessing plants or final disposal preconditioning facilities;
- minimization of the operations for reloading, especially for reloading under water after intermediate dry storage.
- participation of Bulgarian companies.

The existence of the AFR facility in Kozloduy is an advantage which has its place in the different schemes for fuel storage. The facility can be used for intermediate storage, loading and reloading operations, buffer storage capacity, intermediate storage before dry storage after 10 cooling period in the AFR pools the residual heat decrease more than two time compared to 3 y cooling period. .

4. Conclusion

The AFR facility is with limited capacity, it is designed only for VVER-440 fuel although within an year it will be technically possible to transport and store VVER-1000 fuel. The AFR facility has also licensing problems due to increased safety requirements. Work is going on for improvement of the facility in order to comply with the new requirements but nevertheless its operation will only postpone the problem with 3-4 y.

There is very little time for solving the problem of SNF in Kozloduy.

The problem of SNF faces all countries which operate NPPs but each county has developed its national policy - final disposal, reprocessing, deferring the decision. In the last case there are storage facilities for all the fuel which is expected to be generated and also there are allocated funds for the back end.

The developed in the last 10 y technologies for modular dry storage with passive cooling have significant advantage - minimal maintenance, increased level of safety, prolonged investment periods, reduction of the generated heat in time which is considered as a an argument for possible storage term extension. Those advantages of the dry storage modular technologies can help for the solving of the problem of SNF in Kozloduy.

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DETERMINATION OF THE DISTRIBUTION OF HIGH-DISPERSITY HOT PARTICLES FROM NPP BY ACTIVITY USING AUTORADIOGRAPHY

Ts. Bonchev[1], V. Angelov[2], Ts. Andreev[3], N. Todorov[3], P. Andreev[4], I. Kasabov[1]

In our recent publications we adduced data determining the distribution parameters of hot particles by activity (following the logarithmic-normal law) using processing of their images on autoradiograms by means of image analyser. Data referred to hot particles arising from the Chernobyl NPP accident deposited on the surface of air filters[1,2]. In 1992 such particles have been detected in the technological rooms of NPP „Kozloduy" and since then they are subject to our systematic investigations whose first results are published in[3].

The present investigation concerns further improvement of the processing technique of hot particle images on autoradiograms.

1. Equipment and processing technique

Autoradiograms have been obtained through direct contact of the sample with ordinary black - white photographic film whose advantages in such an investigation are given in[1-3].

From the negative a positive on photographic paper with five times augmentation is obtained.

Using iPhoto Deluxe B105A scanner which distinguishes 256 levels of blackening the image is led in PC IBM486.

Images of „standard" particles are led in the same way.

Examples of processing of such images are given in Fig. 1 and Fig. 2.

A special software determining the image diameter of any particle in chosen blackening level (equidensite) and then finding the distribution of images in diameter value is developed.

From the standard particles the relation between the image diameter and the activity is determined for every blackening level. In such a way the distribution of the autoradiogram particles in activity is find. Along with this the logarithmic-normal distribution parameters of the particles from the autoradiogram are obtained.
An example of this is given in Fig.3.

2. Some problems in autoradiogram processing and the matter of the obtained ultimate results

As is known the parameters of an image on photographic paper depend on many factors: the photographic paper type, the exposure time, the developer nature, the developing conditions. To obtain enough well results it is necessary the images of samples and standard particles to be developed at the same conditions, in particularly - the photographic materials to be developed at the same time.

The same problems exist in positive images obtaining. Depending on the exposure time in positive obtaining there are many images whose diameters expand with exposure time. At higher density of the hot particles at unity area some of images merge. The developed software allows to distinguish the separate particles to certain extent even when their images are partially merged.

Varying the exposure time in positive image obtaining we can get information on a long range of activities. A measure for the result reality is that the number of the particles with

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[3]NPP „Kozloduy"
[4]MICROLAB 91, Sofia

the same activity could not depend on that exposure time. Here we purposely give a very „hard case" of an autoradiogram consisting of a great number of images and the results of their processing concerning the above mentioned condition, Fig. 4 (a-d) and Fig. 5. For the same images there is a satisfactory coincidence of the logarithmic-normal distribution parameters of the hot particles by activity at different exposure times (Table 1).

All of this shows the suitability of the worked out technique and software for obtaining quantitative results processing the images of complicated hot particle autoradiograms.

The standard particles consist only $^{90}Sr+^{90}Y$ while the NPP hot particles have a complex radionuclide composition. The conditions of „exposure" on the photographic material and the „exposure" on the lung are close because of which the NPP hot particle activities can be determined as „strontium-equivalent". This gives a possibility to forecast their impact on lungs of human body on the base of an developed model of the impact of particles consisting $^{90}Sr+^{90}Y$ [4].

3. Investigation of technological rooms in NPP „Kozloduy" contaminated with high-dispersity hot particles

Using the above mentioned technique more than 60 autoradigrams of samples from the NPP „Kozloduy" technological rooms are examined. For every separate case the results are given as is shown in Fig. 6.

The analysis of the results obtained shows that the activity mean values and, which is more important, the dispersity values in logarithmic-normal distribution, σ_L, are relatively close. This gives the possibility to average the assessments for the hot particle parameters from different autoradiograms (samples) of the NPP technological rooms and special rooms containing the radioactive waste The results are given in Table 2.

4. Determining the specific activity of hot particles, their distribution by weight, assessment of precipitation time on surfaces and so on

From the images of separate hot particles obtained by means of scanning electron microscope and determination of their activity the assessment of hot particle specific activity is obtained in Bq/cm^3. After the particle composition identification by means of electron microprobe their specific activity is determined in Bq/g. The specific activity concerning the alpha-radiation is obtained determining the radiation layer thickness from the alpha-spectrum „dilution". There are too many results compared but we can not give them here. In the final analysis most reliable are the values in Table 3.

On the base of these results and using the logarithmic-normal distribution law as well as for particle precipitation depending on their Stokes radius assessment of the hot particle precipitation time on the surface in a technological room is given. Using the recommendations on the assessment of the probability for retention of hot particles in lungs shown in[5] there are calculation of the expected activity in the lungs of people in NPP „Kozloduy".

Conclusion

On the base of the developed techniques and the results from the investigation in NPP „Kozloduy" up to now there is a possibility to obtain sufficiently precise assessments for the parameters of high-dispersity hot particles in NPP and hence to provide for their impact on the NPP staff. On the same base it is possible to be recommended grounded measures for staff protection.

Acknowledgements

We take the advantage of the opportunity to express our gratitude to IAEA and the NPP „Kozloduy" administration for the material and moral support for these investigations.

Now a paper for scientific journal of a part of the results obtained up to now is in progress.

Fig. 1 Three-dimensional picture of an autoradiogram of a hot „standard" particle

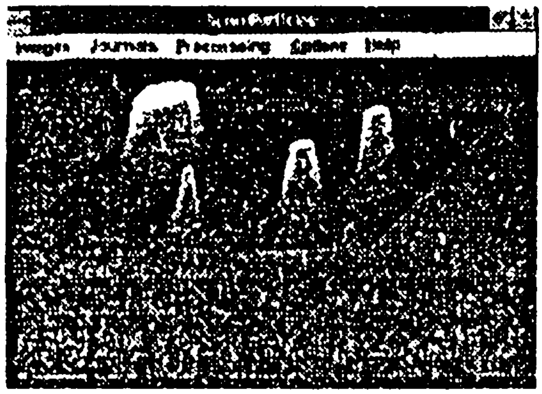


Fig. 2 Three-dimensional picture of an autoradiogram of a sample from NPP „Kozloduy"

Fig. 3 An example of result processing of an autoradiogram

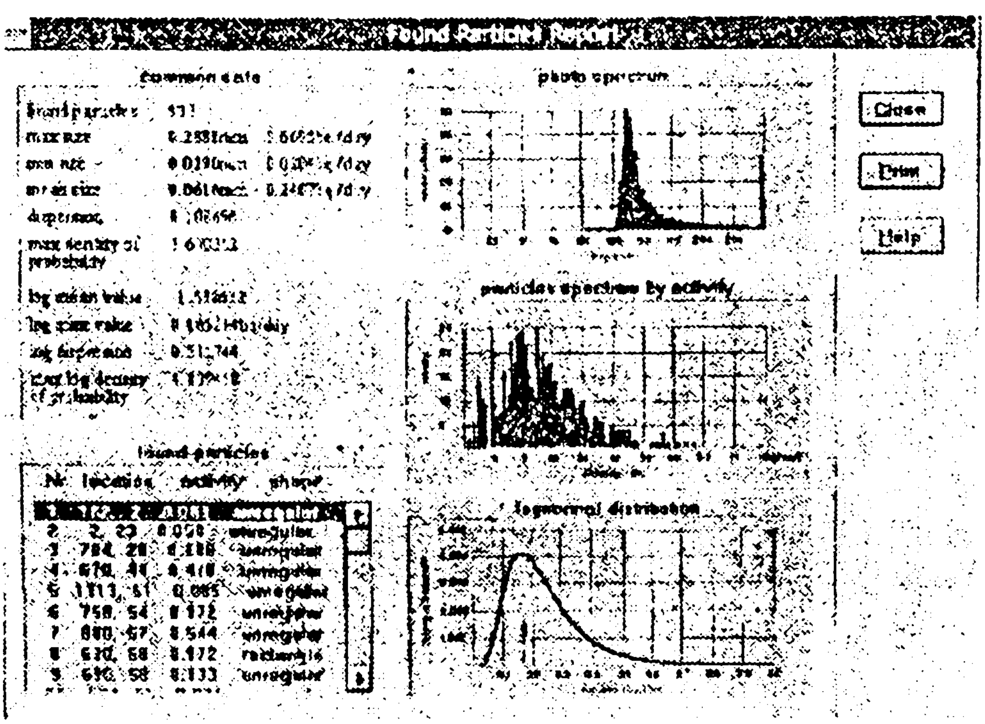


Fig. 4 Positive image of an autoradiogram at different exposure times

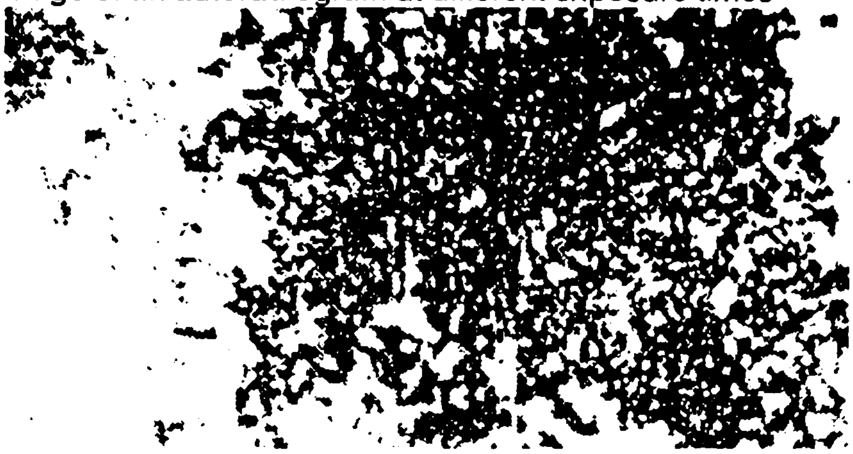


Fig. 4(a) Exposure 2 s

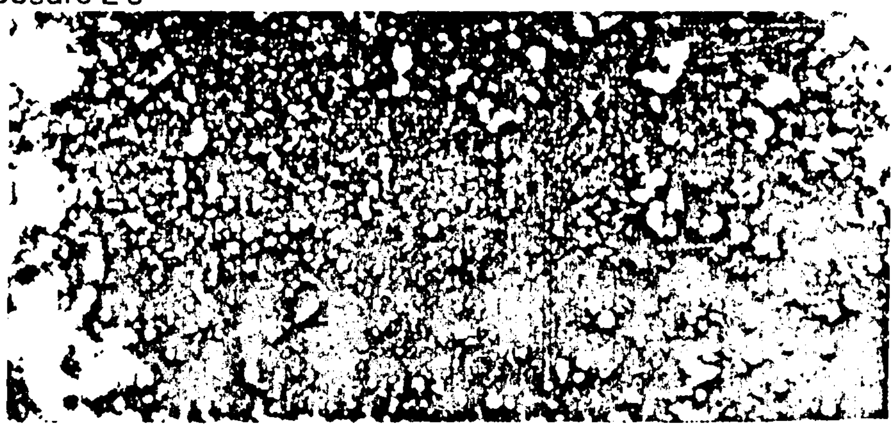


Fig.4 (b) Exposure 3 s

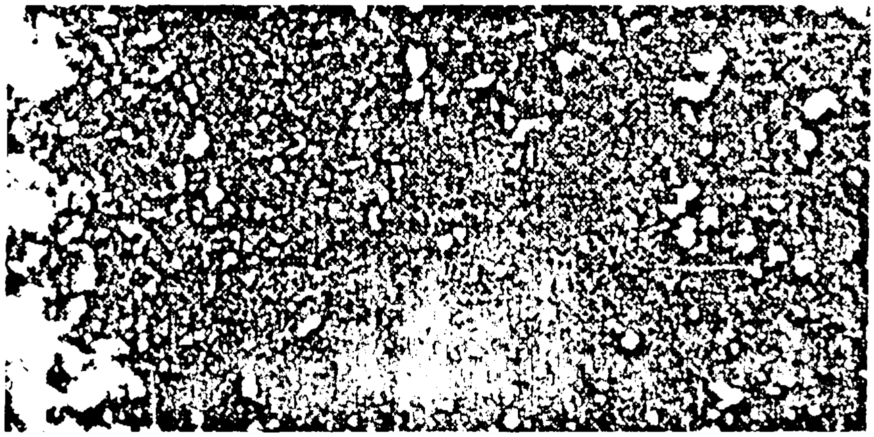


Fig. 4(c) Exposure 4s

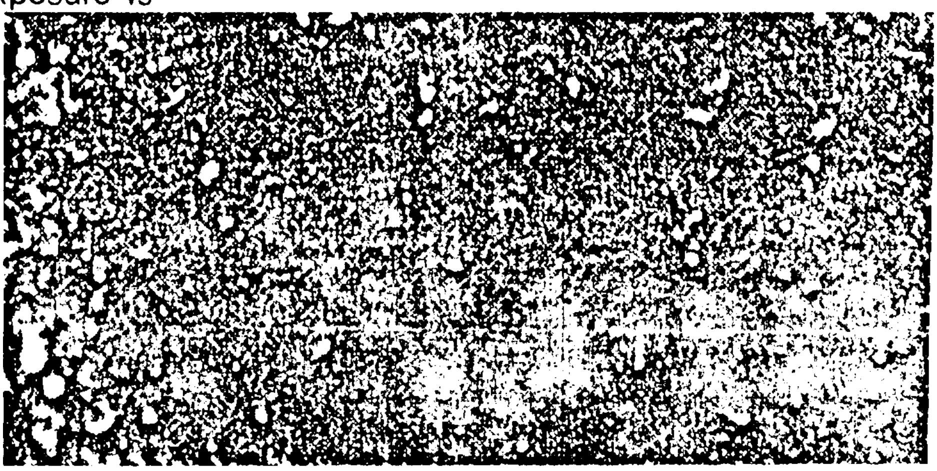


Fig. 4(d) Exposure 6s

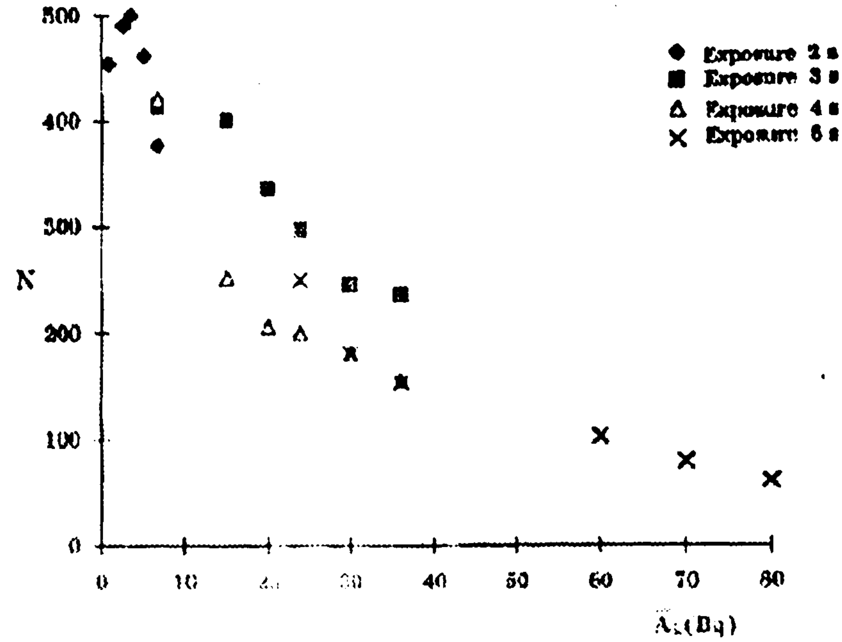
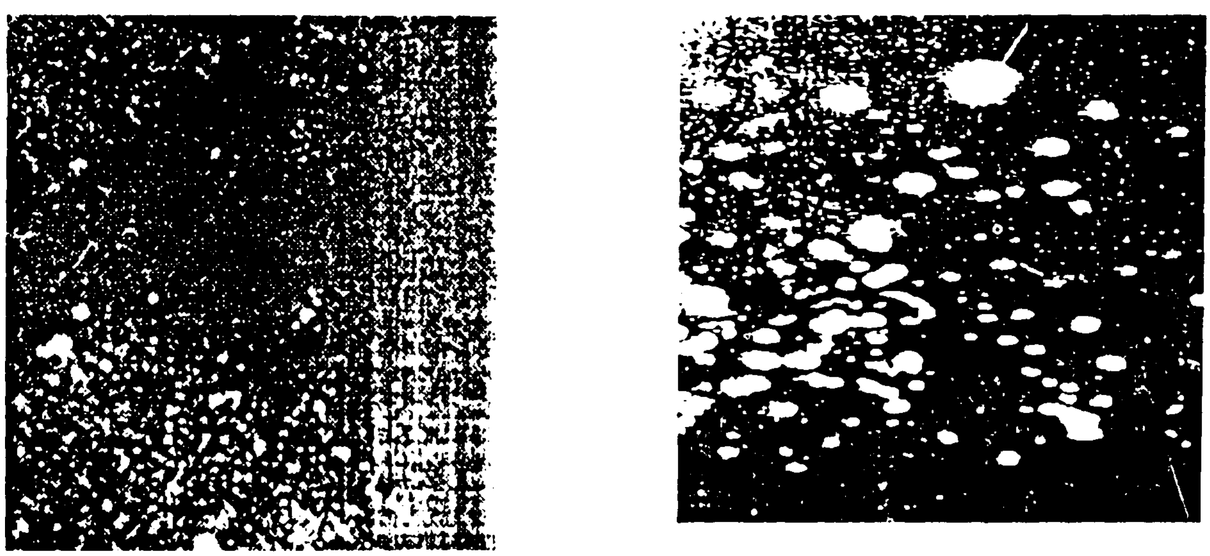


Fig. 5 Relation between the number of the „accounted" particles, N, and the value of $\overline{A_L}$

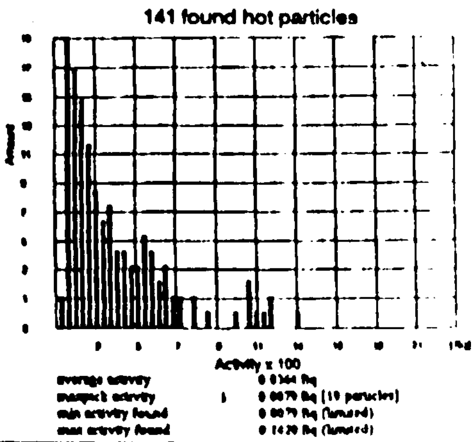
Fig. 6 Examples of particular results for NPP „Kozloduy"

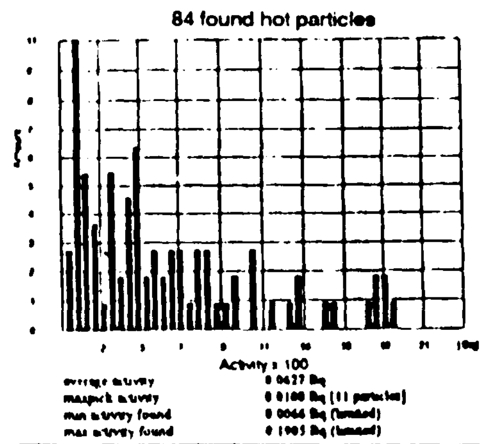


SK-I, 24.03.1993
No.087 and No.088
From the floor of BK008

SK-I, 24.03.1993
No.089 and No.090
From the pipe of BK008

Exposure 10 d; Magnif. x3	Exposure 10 d; Magnif. x3





$\overline{\ln A}$	-3.578	$\overline{\ln A}$	-3.114
σ_L	0.720	σ_L	0.882
$A_{L\max}$	0.017 Bq	$A_{L\max}$	0.020 Bq
$\overline{A_L}$	0.036 Bq	$\overline{A_L}$	0.066 Bq

Table 1 Values of N, $\overline{\ln A}$ and σ_L corresponding to the same value $\overline{A_L}$ for the autoradiograms shown in Fig.4(a-d)

$\overline{A_L}$ (Bq)	4(a)			4(b)			4(c)			4(d)		
	N	$\overline{\ln A}$	σ_L	N	$\overline{\ln A}$	σ_L	N	$\overline{\ln A}$	σ_L	N	$\overline{\ln A}$	σ_L
0.9	453	-0.59	0.95	609	-0.62	1.02						
2.6	490	0.15	1.28	573	0.08	1.33	513	0.30	1.13			
3.5	500	0.47	1.28	471	0.40	1.30						
5.0	460	0.78	1.30	439	0.77	1.30						
6.8	378	1.09	1.29	414	1.07	1.29	420	1.04	1.33			
15				401	1.66	1.45	252	1.98	1.24			
20				337	2.10	1.30	206	2.30	1.20			
24				297	2.50	1.20	200	2.45	1.17	250	2.40	1.30
30				246	3.00	0.90	180	2.70	1.20	180	2.70	1.20
36				236	3.40	0.66	154	3.00	1.10	151	2.90	1.18
60										103	3.60	0.97
70										80	4.00	0.62
80										62	4.30	0.37

Table 2 $\overline{A_L}$ and σ_L averaged parameters for different reactor units and special rooms (SK) containing waste; n - number of examined samples

Object	n	$\overline{A_L}$ [Bq]	σ_L	
Reactor unit I	13	0.097	0.77±0.11	
Reactor unit II	11	0.107	0.92±0.12	
Reactor unit III	9	0.120	0.88±0.20	
Reactor unit IV	4	0.171	0.87±0.20	
Average	37	0.114	0.85	$\overline{\ln A}=-2.53$
SK-1	4	0.042	0.803± 0.057	
SK-2	5	0.037	0.811±0.066	
Average	9	0.039	0.807	$\overline{\ln A}=-3.57$

Table 3 Specific activity, a_v and a_p, of hot particles from NPP „Kozloduy"

Emitter	a_v [Bq/cm^3]	a_p [Bq/g]
^{60}Co	1×10^9	2×10^8
^{54}Mn	1×10^8	2×10^7
α-emitters	$.3\times10^5 \div 2.8\times10^4$ Bq/g	

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ОПРЕДЕЛЕНИЕ СОДЕРЖАНИЯ ФОСФОРА, МЕДИ, НИКЕЛЯ, ХРОМА, ВАНАДИЯ И ТИТАНА В СТРУЖКАХ 4 ШВА I БЛОКА АЭС КОЗЛОДУЙ

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Для оценки хрупкости металла корпусов реакторов необходимо знать содержание в стали примесей фосфора, меди, никеля и других элементов. Для анализа образцов были использованы надежные и хорошо отработанные методы. Одним из методов определения фосфора, как наиболее критического елемента, было его спектрофотометрическое определение в соответствии с ГОСТом (1,2). Остальные елементы определяли с помощьЮ эмиссионной спектрометрии с плазменным возбуждением (AES-ICP).

В 1989 году после отжига корпуса реактора первого блока АЭС Козлодуй были взяты пробы стали 4 шва корпуса. Стружки были отобраны в трёх местах шва под углом 120°. Образцы были взяты с различной глубины.

Приготовление образцов

Стружки тщательно промывались водой, с добавкой небольшого количества детерегента, в ультразвуковой ванне. После ванны стружки последовательно промывались дистиллированной водои и спиртом. Затем сушились и взвешивались с точностьЮ до 4 знака. Масса образцов представлена в Таблице 1.

Таблица 1. Масса образцов стали 4 шва корпуса реактора I блока.

N° ОБРАЗЦА	МЕСТО ОТБОРА	ГЛУБИНА ОТБОРА, ММ	МАССА, Г
6	2	3.4 - 4.1	6.3414
8	2	4.1 - 4.8	3.7396
9	2	4.8 - 5.5	0.0387
10	3	4.0 - 4.8	6.8513

Масса образца N° 9 недостаточна для надежного определения фосфора. Поэтому с помощьЮ AES-ICP в нем было определено содержание всех упомянутых примесей кроме фосфора.

Растворение образцов и приготовление растворов

Образцы растворяли в 40 мл смеси соляной и азотной кислот в соотношении 3:1 при нагревании до 80° в 250 мл стакане покрытом часовым стеклом. После растворения раствор выпаривали до влажных солей. Далее добавляли 10 мл концентрированной соляной кислоты и снова выпаривали. Процедуру повторяли три раза. Остаток растворяли в 6М соляной кислоте и фильтровали через бумажный фильтр „синяя лента". Фильтрат количественно переносили в мерительную колбу и добавляли до марки 6М солянуЮ кислоту. Эти растворы были исходными для приготовления рабочих растворов. Концентрации и количества используемых исходных и рабочих растворов представлены в Таблице 2.

Определение фосфора

Спектрофотометрическое определение фосфора в водной фазе (Вариант 2.1.3. по ГОСТу)

Метод основан на образовании желтой фосфорно-молибденовой гетерополикислоты ($H_3[P(Mo_{12}O_{40})].nH_2O$) и ее последующим восстановлением ионами двухвалентного железа в присутствии гидрохлорида гидроксиламина до синего комплекса. Комплекс устойчив не менее 1.5 ч. Оптическую плотность раствора измеряли на спектрофотометре при длине волны 830 нм. Фосфор (III) предварительно окисляют до фосфора (V) раствором перманганата калия. Мышьяк отгоняют в виде $AsCl_3$.

Экстракционное выделение фосфора и спектрофотометрическое определение в органической фазе (Вариант 3 по ГОСТу).

Метод основан на экстракции фосфора изобутиловым спиртом из хлорной кислоты. Образующийся желтый комплекс в органической фазе восстанавливают хлористым оловом до синего комплекса и измеряют оптическую плотность при длине волны 725 нм. Комплекс устойив в продолжение 3 часов. Мышьяк предварительно отгоняют в виде $AsCl_3$. Хром отгоняют в виде хлористого хромила. В обоих случаях был использован спектрофотометр Spekol 11, производства Carl Ziess, Jena.

Определение фосфора с помощью эмисионной спектрометрии с плазменным возбуждением (AES-ICP).

Растворенный образец разбавляется минеральной кислотой (соляной или азотной) и подается в плазменную горелку. Здесь атомы возбуждаются, испуская характеристическое излучение в оптическом диапазоне. По интенсивности спектральных линий судят о количестве элемента. Для сравнения используются растворы известных концентраций. Был использован спектрометр фирмы ARL (США) модели 3520.

Приготовление стандартных растворов для определения фосфора

Приготавливали два стандартных раствора. Один в соответствии с ГОСТом из K_2HPO_4 и другой из H_3PO_4 в соответствии с рекомендациями NIST, США (бывшее Национальное Бюро Стандартов, NBS)(3). В рамках допустимых отклонений оба стандарта дали одинаковые результаты. Для учета влияния железной матрицы при определении содержания фосфора с помощью AES-ICP, в стандарт добавляли чистое железо до концентрации 20 мг/мл.

Полученные результаты и их обсуждение

Построение градуировочных графиков.

Градуировочная прямая описивается уравнением:

$$E = к.C \tag{1}$$

где: E - оптическая плотность раствора,
C - концентрация фосфора в растворе,
к - коэффициент

На рис.1 представлен градуировочный график для определения фосфора в водной фазе. Коэффициент к в данном случае равен 0.1360 ± 0.0007.

В' Таблице 3 представлены результаты анализа, полученные при спектрометрировании водной фазы.

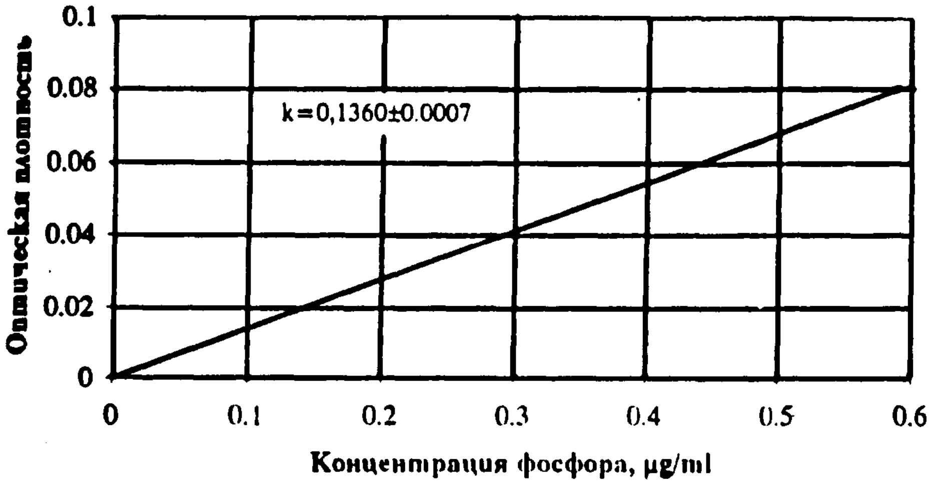

Рис.1. Градуировочный график для спектрофотометрорования водной фазы.

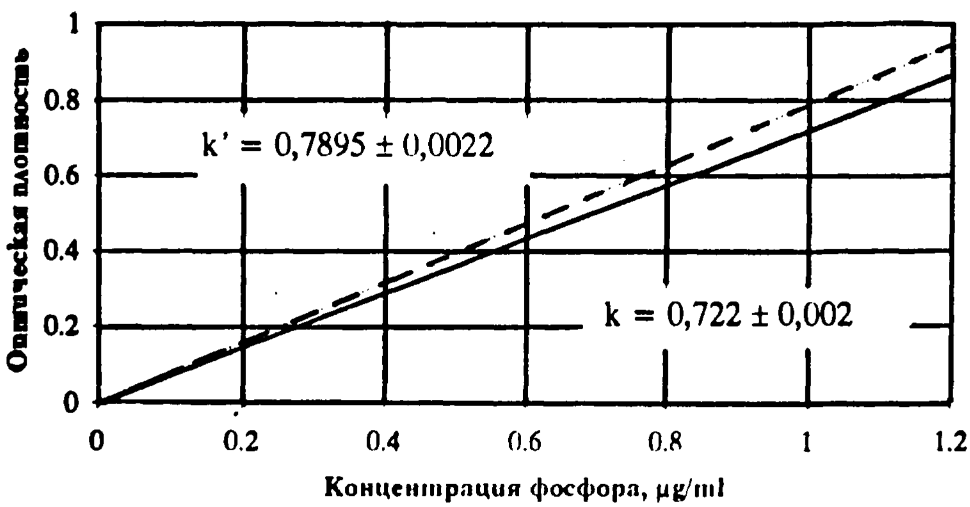

Рис.2. Градуировочный график для спектрофотометрорования органичной фазы.

На рис. 2 представлен градуировочный график для спектрометрирования органической фазы. Коэффициент к в данном случае равен 0.722 ± 0.002. Используя это значение коэффициента к били определены концентрации фосфора. Эти результаты представлены в Таблице 4.

Результаты завышены в сравнение с полученными при спектрометрирова-нии водной фазы. Это завышение определялось двумя факторами:

-во-первых, процедура приготовления растворов для получения градуировочного графика по ГОСТ 12347-77 (п.3.3.3.) не обеспечивает той же кислотности среды, что и у образцов. При одинаковой кислотности градуировочных растворов и образцов величина коэффициента к увеличилась на 4%;

во-вторых, в стали предполагается содержание ванадия до 0,15%. При добавлении в градуировочные растворы ванадия до концентрации 0,15% коыффициент увеличился еще на 6%.

С учетом этих поправок, его новое значение равно 0,7895 ± 0,0022.

Результаты рассчитанные с этим коэффициентом, как и результаты от спектрофотометрирования водной фазы и эмиссионного спектрального анализа (AES-ICP) представлены в сводной Таблице 5

Представленные результаты идентичны в рамках указанных погрешностей определенных с P=0.95 доверительным интервалом, включая и погрешности градуировки. Следует отметить, что непосредственное и8спользование процедуры ГОСТа (12347-77) не обеспечивает необходимой точности. Контроль кислотности и поправка на содержание ванадия значительно улучшают качество результатов. Точность спектрального анализа (ICP) можно повысить удалив железную матрицу.

Определение содержания меди, никеля, хрома, ванадия и титана

По времени определение указанных элементов производилось после определения в стружках фосфора. Использовались остаточные растворы, что естественно ограничивало возможности анализа. В Таблице 6 приведены длины волн аналитических спектральных линий определяемых элементов.

Для каждого элемента были приготовлены по две серии стандартных растворов. Одна серия - чистые растворы. Вторая, для оценки влияния железной матрицы, содержала добавку железа с концентрацией 2,5 mg/ml. Концентрации стандартных растворов представлены в Таблице 7.

В Таблице 8 представлены результаты определения меди, хрома, никеля, ванадия и титана в стружках.

Все полученные результаты использованы фля расчета хрупкости металла корпуса.

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Таблица 2. Концентрации и количества исходных и рабочих растворов для определения содержания фосфора.

N° пробы	Исходные растворы		Рабочие растворы					
			определение фосфора				опред. других елементов	
			спектрометрирование водной фазы		спектрометрирование орг. фазы			
	объем [ml]	концентрация стали mg/ml	объем ml	кол-во стали, g	объем ml	кол-во g	объем ml	концентрация стали, mg/ml
6	200	32,2	10	0,3216	20	0,6431	50	19,3
8	100	37,4	10	0,3740	-	-	50	22,4
9	2	19,4	-	-	-	-	2	19,4
10	200	34,3	10	0,3426	20	0,6851	50	20,6

Таблица 8. Содержание меди, никеля, хрома, ванадия и титана, весовые %

Проба	Ni		Cr		V		Ti		Cu	
	определено	паспорт (4)	определено	паспорт (4)	определено	паспорт (4)	определено	паспорт (4)	определено	паспорт (4)
6	0,260 ±0,06	0,29	1,51 ±0,06	1,58	0,103 ±0,01	0,15	0,002 ±0,0006		0,133 ±0,004	0,12
8	0,249 ±0,06	0,29	1,47 ±0,06	1,58	0,105 ±0,011	0,15	0,002 ±0,0006		0,115 ±0,004	0,12
9	-	0,29	-	1,58	-	0,15	-	-	0,129 ±0,004	0,12
10	0,249 ±0,06	0,29	1,54 ±0,06	1,58	0,106 ±0,011	0,15	0,0029 ±0,0006		0,111 ±0,003	0,12

Таблица 3. Результаты определения фосфора при спецтрометрировании водной фазы

N° пробы содержание стали в аликвоте g)	Экстинкция		Содержание фосфора, весовые %
	отдельное определение	средняя	
6 (0,3216)	0,04333 0,0438 0,0432 0,0444 0,0442	0,04370 ±0,00024	0,05000 ошибка определения: ±0,00027 ошибка при градуировке: ±0,00024 ошибка общая (P=0,95): ±0,00086
8 (0,3740)	0,048 0,0506 0,0505 0,0484	0,04938 ±0,0007	0,04859 ошибка определения: ±0,00067 ошибка при градуировке: ±0,00026 ошибка общая (P=0,95): ±0,0022
10 (0,34260)	0,0449 0,0428 0,0433 0,0415	0,0431 ±0,0007	0,04629 ошибка определения: ±0,00075 ошибка при градуировке: ±0,00024 ошибка общая (P=0,95): ±0,0024

Таблица 4. Содержание фосфора при спектрофотометрировании органической фазы.

Проба No	Содержание фосфора, весовые %
6	0.0573 ± 0.0008
10	0.0530 ± 0.0007

Таблица 5. Содержание фосфора в стружках, весовые %.

Проба No	Содержание фосфора, весовые %		
	Спектрофотометрия		Эмисионная спектрометрия (ICP)
	водная фаза	органическая фаза	
6	0.05±0.0008	0.0521±0.0008	0,052±0,005
8	0.0485±0.0006		0,041±0,004
9			0,034±0,003
10	0.0462±0.0007	0.0482±0.0007	0,044±0,004

Таблица 6. Длины волн аналитических спектральных линий, nm.

Ni, λ	Cr, λ	Cu, λ	V, λ	Ti, λ
231,684	267,716	224,70	290,882	334,940

Таблица 7. Концентрации стандартных растворов, µg/ml

никель	хром	медь	ванадий	титан
0,45	13.58	0,22	0.49	0,25
2,23	15,35	0,14	2,00	0,77
4,46	27,16	1,11	4,03	1,25
7,44	46,08	2,23	5,00	2,51
11,16	67,09	4,46	8.06	5,05

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Conversion of Highly Enriched Uranium in Thorium-232 Based Oxide Fuel for Light Water Reactors: MOX-T Fuel

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Abstract
The possibility to use highly enriched uranium from military inventories in mixed oxide fuel U-235 and Th-232 for light water reactors has been considered. It is shown that although the fuel conversion coefficient to U-233 is expected to be less than 1, the proposed fuel has several important advantages which will result in reduction of the cost of the nuclear fuel cycle. Due to fuel generation the initial enrichment is expected to be less by ~1% for the same energy production and also no long living actinides are generated which greatly reduces the cost of fuel disposal or reprocessing.

Introduction
A problem which has emerged in the recent years is the conversion of the existing stockpiles of weapon-grade highly enriched uranium (HEU). According to the estimations the 50 000 warheads in the United States and the Russian arsenals contain some 1000 t of HEU and 220 t of plutonium [1]. The most straightforward solution for the HEU is blending it with natural uranium. The argument in favor of the blending is that the process is irreversible and such a technology helps the observation of the Non-Proliferation Treaty. An argument against this solution is that a lot of energy has been used for the enrichment - the ratio of the separation work units for HEU and uranium with ~ 4% enrichment is roughly 60 [2] and mixing the enriched uranium with U-238 has to be considered as the last option. The enrichment should result in improvement of the quality of tne existing nuclear fuel, creation of new types of fuel or improvement of the fuel cycle.

Another problem of the contemporary nuclear fuel cycle is the inevitable generation of Pu and other higher long living actinides. The long-living actinides need to be conditioned as high level radioactive waste (HLRW) in case of fuel reprocessing and the separated plutonium reused in mixed oxide (MOX) fuel, or it has to be safely stored. For the rest of the actinides - americium and the even isotopes of plutonium extensive research programs exist for transmutation (fissioning) with fast neutrons. In the case of final disposal of spent nuclear fuel the actinides are the main source of heat generation after 150 y of cooling time and the main concern for estimation of the repository status after thousands of years.

The aim of the present paper is to discuss the feasibility of an idea for conversion of HEU without mixing with natural uranium and the utilization of Pu in a fuel in which no new plutonium is generated.

The fuel for conversion of HEU and Pu utilization - principles.

The proposition is to use U-235 (HEU) for a fissile isotope and Th-232 as a non-fissile isotope in a mixed oxide with thorium fuel for light water reactors. For most of the LWRs the percentage of the fissile isotopes is approximately 4% in the form of dioxide and for the rest 96% ThO_2 is proposed. Plutonium can also be used in the proposed fuel as a mixture with U-235 [3].

The main advantage is that weapon-grade HEU can be utilized by reversible blending with Th, a mono-isotope element which is at least as abundant as uranium. The possibility for reverse extraction of U-235 and the generated U-233 is also a weak point of the proposed fuel because of safeguard reasons.

The second advantage of the proposed thorium based fuel is that no Pu-239 and other long living actinides are generated. If the enrichment of the HEU is 93%, the presence of only 0.35% U-238 for a fuel of approximately 4.5% enrichment means that the generated Pu-239 and other higher actinides are expected to be 400 - 500 less than in conventional fuel. That is a great advantage in case of reprocessing of the fuel or final disposal. After 10 y of cooling in intermediate storage facilities the residual heat will be generated mainly by Cs-137, Sr-90 and some 10-15 g/t HM of trans U-238 actinides instead of 5 - 8 kg/t HM of actinides.

The third advantage is that the generated U-233 after neutron capture by Th-232 is a very good fissile isotope which will be generated at 3 times higher rate than Pu-239 from U-238. The high rate of production of U-233 and the high value of secondary neutrons per one absorbed neutron of U-233 can make the fuel cycle with very high fuel conversion coefficient or even with breeding of fuel under special conditions.

The fourth advantage is that in case of fuel reprocessing all fissile material (U-233 +U-235) can be chemically extracted and no U-235 will be lost.

Up to now the thorium cycle has been considered only for breeding or obtaining a fuel cycle with fuel conversion coefficient greater than 1. Most of the research has been done for a thorium cycle with fast breeders or crossed cycles - breeders and reactors with thermal neutrons. The proposed fuel cycle is expected to be with fuel conversion coefficient less than 1 but with much higher coefficient than that of LWRs with U-235 +U-238 fuel. The additional energy generation will result in cheaper nuclear energy.

Estimation of the expected propert es of the mixed with thorium fuel - cross-sections, generation chains, delayed neutrons.

There is enough experience in the utilization of Th-232 and generation of U-233. The first experimental data have been obtained in the BWR Indian Point. The experiments showed that except in breeders high conversion ratio or breeding can be achieved in thermal reactors with low capture cross-section in the moderator and coolant - e.g HTGR, AVR, CANDU. The fuel elements of HTGRs contain two types of micro spheres - the first type is either U-233 or HEU U-235 in the form of UC covered with carbon and silicon carbide and the second type is made of thorium dioxide covered with carbon [4].
The cross-sections for neutron capture and scattering of thermal neutrons of Th-232 and U-238 are comparable but since the capture cross-section of Th-232 is ~ 3 times greater than the capture cross-section of U-238, the rate of generation of the fissile isotope U-233 will be 3 times greater than the generation of Pu-239 in conventional fuel.
The generation chain of U-233 is very similar to that of Pu-239:

$$\text{Th-232} + n \rightarrow \text{Th-233} \xrightarrow{22.1\ \text{min}} \text{Pa-233} \xrightarrow{27\ \text{d}} \text{U-233}$$

$$\text{Pa-233} + n \rightarrow \text{Pa-234} \xrightarrow[1.2\ \text{min}]{} \text{U-234}$$

The 27 d isotope Pa-233 has a relatively high capture cross-section for thermal neutrons (21 b [5]), or a small fraction of neutrons will be lost until Pa-233 decays to U-233. In terms of reactivity the negative contribution from Pa-233 will be compensated by the lower cumulative yield of Xe-135. The ratio of the cumulative yield of Xe-135 from U-235 and U-233 is 1.41 [6].

The values for the cross-sections for capture and fission with thermal neutrons of U-233, U-235 and Pu-239 are very close or no great changes in the concentration of the fissile isotopes are necessary.
The relatively low value of the capture cross-section of U-233 increases the effective value of the number of secondary neutrons (4) per one absorbed neutron (2.28 [4]). That high value also increases the possibility for fuel breeding.
The approximate calculations for a homogenized core of VVER-1000 type for a 4.4% U-235 + Th show that the burnup can be increased by approximately 1/3 for the usual reloading schemes because of generation and burning of U-233. The additional positive reactivity is generated during the operation and there is no need for compensation of the excess reactivity in the "fresh" core. For a burnup of ~40 MW.d/tU the initial enrichment of U can be reduced by ~1%, or the U-235+Th fuel reduces the price of the nuclear fuel.

The value of the fraction of delayed neutrons of the generated from thorium U-233 is greater but close to that of Pu-239, or fuel of only Th + U-233 requires a modification of the control system. The presently assumed ratios of U-235 and Pu-239 in the MOX fuel are acceptable for the proposed thorium fuel with fissile isotopes U-235 and U-233.

The generated U-233 and the residual U-235 can be chemically separated from the rest of the thorium fuel. Except fission fragments, the long living isotope U-232 (T1/2 =72 y) is generated by (n,2n) reaction:

Th-232(n,2n)Th-231---------> Pa-231 + n--------> Pa-232----->U-232

U-232 and its daughters are radioactive but the decay periods of the daughters are shorter. The daughter isotope with the longest decay period is Th-228 (1.9 y).

Technological compatibility of the thorium based fuel.
The most important technological difference of the thorium dioxide compared to uranium dioxide is the specific density [7]. The obtained density of UO_2 is less than the theoretical one by 0.4 - 0.6 g/cm^3 and that will be the case most probably with ThO_2 too. The lower density means that approximately 10% less fuel will be loaded in the reactor core. The higher melting point of ThO_2 is an advantage of the proposed fuel.

Except as a mixture of UO_2 and ThO_2 the uranium-thorium fuel can be made of physically separated substances - e.g. UO_2 coating over ThO_2 cylinder.

Conclusion
The advantages of the use of HEU in LWRs in mixed U-235 - Th fuel are:
- no generation of long living plutonium and americium isotopes. In the case of reprocessing the HLRW will contain only fission fragments and U, in the case of final disposal of the spent nuclear fuel the repositories should be designed for shorter periods (e.g. 500y);
- the high conversion ratio of Th extends the expected burnup (by approx. 1/3) without higher initial enrichment. The same initial enrichment simplifies the problem for compensation of the excess reactivity in the beginning with burnable poison and boric acid;
- the high conversion ratio of Th makes possible the utilization of fuel with less initial enrichment (by approx. 1/3) for the same burnup. Thus less excess reactivity has to be compensated after reloading.
- in case of fuel reprocessing all fissile material (U-235+U-233) can be chemically extracted.

The proposed fuel or the Mixed OXide with Thorium - MOX-T fuel, actually extends the limits of depletion of fissile isotopes since the use of Th can slow down the rate of burning of uranium. The total amount of 1000 t HEU can supply fuel for 3.5 - 4 years for the reactors all over the world. The projected rate of dismantling is 2000 USA warheads annually and 1500-2000 Russian warheads, or 30 to 40 t of HEU are expected to be available per year. Depending on the type of the reactor, 50 to 80 reactors can be refueled with uranium from HEU conversion.
The proposed fuel can utilize also weapon-grade plutonium and plutonium from reprocessed spent nuclear fuel in a U + Pu + Th composition without generation of new plutonium.

If large quantities of HEU from nuclear warheads become available that will be a very unique possibility to start a large scale fuel conversion of Th within the clean fuel cycle with U in light water reactors and to improve the economics of the nuclear energy.
The further work on the problem includes calculation of optimal loading and reloading schemes, theoretical calculation of the thermal properties of fuel pins with U-235+Th fuel and which is the most important and the most difficult problem - manufacturing of several test fuel assemblies and observation of their behavior in a reactor core.

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ENVIRONMENTAL MONITORING OF NPP „KOZLODUY": 20 YEAR EXPERIENCE

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1. Introduction

The industrial use of nuclear energy raised a new, global problem regarding environment - the environmental contamination with artificial radionuclides. Radionuclides emission and migration, their accumulation and behaviour in the different parts of the food chain, as well as their biological impact are processes of considerable interest.

Nowadays, a subsequent rearrangement of the world fuel-energy balance is taking place. As apart of it, nuclear energy is assumed as the only alternative of the up to now power generation, based on fossil-fuels. The development of the society depends on the energy production means, and obtaining energy on its turn affects the environment and the population. In this way the problems of energy and ecology are linked together. The main consideration concerning the development of nuclear power generation is basically related to the social acceptance of the 'cost-benefit' criterion, which is a basic feature of the social development level. A basic requirement to the safe operation of a nuclear power equipment is the possibly most full isolation of the contamination sources and restriction of artificial radionuclides releases to the environment.

The development and operating of an effective radiation monitoring system, which is a compulsory condition during the operation of nuclear power facility, provides obtaining of operational information on the current values, as well as on the trends of the radiological conditions. This system includes five relative subsystems, which are not clearly distinguished:

- technological radiation monitoring;
- individual dosimetric control;
- control of the non proliferation of radioactive contamination;
- environmental radiation monitoring;
- radiation control in case of an accident.

Each one of the subsystems covers a specific scope of the radiation monitoring. It includes different types of measurements, analysis and assessments. Finally it aims at protection of life and health of the plant personnel and the general population, as well as at the restriction (within the range of the permissible limits) of the environmental radioactive contamination.

The purpose of this paper is to present the radiation monitoring system and the assessment of the impact of NPP 'Kozloduy' to the environment.

II. Environmental surveillance

The main goal of the performed radiation monitoring is to avoid violations of the permitted dose limits, thus providing health protection of the general population.

1. Legislation, rules and regulations

The Committee of the Use of Atomic Energy for Peaceful Purposes (CUAEPP) is the main authority responsible of carrying out an unified state policy in the area of the use of nuclear energy under the Law of the Use of Atomic Energy for Peaceful Purposes [1]. It is written in Chapter 1, Article 3 that 'protection of life and health of the population and of the environment is of primary importance than the economical and other social demands'. The Ministry of Health (MH) and Ministry of the Environment (ME), in compliance with the same law, are conceded to perform specialised surveillance within the frame of their responsibilities. Similar texts are available in the Law of Public Health [2] and in the Law of Environmental Protection [3]. The surveillance activities of CUAEPP, MH and ME are assigned to their administrative structure as follows: The Inspection for Safe Use of Atomic Energy (ISUAE), the radiation departments of the Inspection of Hygiene and Epidemiology (IHE) and the environmental radiation laboratories of the regional inspections for environmental protection RIEP (application 2). The above control is carried out in compliance with the Basic Regulations for Radiation Protection (BRRP-92) [4] and the current rules, directives and instructions published by the responsible authorities (application 3).

For the general population in Republic of Bulgaria (e.g. Category B), the internal or external exposure level of the effective dose is 1 mSv/a and the organ and system limits are 10 mSv/a (excluding skin, for which the limit is 50 mSv/a). On the base of the above dose limits, secondary levels are derived for the annual income and for the permissible concentrations of definite radionuclides in air, drinking water and foodstuff. These values do not differ substantially from the ones recommended by ICRP and other international organisations.

2. Pre-start radiation monitoring

The aim of the pre-start monitoring is to define the actual radiological conditions prior the plant start and the assessment of the radiation impact of the already existing natural sources. The collected data are used as a reference to detect any further changes when the plant is in operation. The pre-start monitoring allows to determine the location of the control points for measurement and sampling, as well as to optimise the control programme and the analytical methods to be used during the plant operation. The pre-start monitoring on NPP „Kozloduy" had been conducted for the period 1968-1974. Detailed description of the carried out research work and the obtained results are presented in [5, 6].

3. Radiation monitoring during normal operation of NPP „Kozloduy"

For the purpose of avoiding and assessment of Kozloduy NPP's impact to the population and environment, the following three areas of control were established around the plant:

- radiation protected area 3 km from the stack of units I and II;
- radiation controlled area 12 km;
- surveillance area 100 km.

There are 36 control points in the 100 km surveillance area set up around NPP, where the sampling and the radiation monitoring is carried out. 33 of these are located within the 12 km radiation controlled area and the rest three are in the cities of Lom, Pleven and Berckovitza respectively. The control points distribution is shown on Figure 1. The control points are divided in two groups according to their functional purpose:

- Points of type „A". The radioactivity of aerosols is controlled for this group of points (aspiration equipment with a replaceable filter), samples of air sediments, soil and grass are collected too. The dose rate is also periodically measured at this type of control points and TLDs are mounted for the annual dose estimation. The number of control points „A" is 10.

- Points of type „B". Samples of air sediments, soil and grass are taken at this type of control points. The gamma dose rate is measured periodically and TLDs are mounted. Besides, TLDs also have been mounted along the fence of NPP, since 1993.

Other types of sampling (water, milk, meat, fish, etc.) are done off-site the above mentioned control points.

The radiation environmental monitoring programme was written in compliance with the following principles:

- measurements and sampling to be carried out predominantly at unfavourable points: for example the dose rate is measured along the plant fence; samples as aerosols, soil, grass etc. are taken from regions of the predominant wind direction;
- parallel measurements and sampling to be carried out also in points not affected by NPP „Kozloduy": for example, water and bottom sediments from the Danube river are taken from points upstream and downstream the plant;
- food products taken for analysis ought to be typical for the region of NPP - in our case these are mostly grain (i.e. wheat, crop, barley and sunflower);
- the controlled radionuclides should be attributed to the plant technology;
- MDA must be low enough, comparable to the global fallout;

The radiation environmental monitoring programme was updated in 1993 and was co-ordinated with the Ministry of Health (MH), Ministry of Environment (ME) and with Committee on the Use of Atomic Energy for Peaceful Purposes (CUAEPP).

The measurement equipment used for radioactivity in environmental samples is presented in Table 1. Table 2 presents the nominal values of the minimum detectable activity (MDA).

The radiation monitoring programme as a whole is adequate to the set up tasks and corresponds to the practice established in other countries. A comparison between NPP's 'Kozloduy' sampling scope and those of some other plants in the United States and Canada is presented in Table 3. A similar comparison is done concerning MDA for different types of samples in Table 4.

Together with the monitoring programme conducted by NPP"Kozloduy" a surveillance is carried out by the authorities, namely MH and ME. For example, sampling and measurement of water, bottom sediments and milk [8] are carried out monthly by MH in the regions of Lom, Harletz and Pleven, as ME monitors the dose rate (34 control points) in the range of 30 km around NPP, as well as water samples from the Danube up- and downstream NPP (12 control points) and soil (10 control points).

The effect of NPP 'Kozloduy' on the environment can be estimated on the basis of the recorded values of radiation monitoring data for 1993 [11], compared to the data obtained for the pre-start period [6]. It is reported in Table 5, where no significant changes could be observed. The increase in the activity of Cs-137 is obviously due to the Chernobyl accident from 1986, as its absolute values are of the lowest one, recorded for Bulgarian soil. The slight increase in the values of Cs-137 in bottom sediments from the Danube is probably due to the NPP's operation. By absolute value however, this radioactivity is similar or lower than the one measured in bottom sediments from the rivers Tzibritza and Iskar [21]. For the above mentioned period a certain decrease of the activity of Sr-90 is observed. It is due to the low deposition of this radionuclide following the Chernobyl accident in the north-east of Bulgaria and its evidence results mainly from the global fallout.

Concerning the normal operation of NPP, the release of artificial radionuclides in environment by gaseous and liquid effluents is subjected to continuous radiation monitoring. In respect to this monitoring, secondary level values were derived determining the limits of the released radioactivity. The permissible average 24h limits for NPP 'Kozloduy" are as follows:

Radioactive noble gases	70	Tbq
Long-lived air-born radionuclides	2	Gbq
I-131	1.4	Gbq

The established concentration limit for liquid radioactive effluents is 11.1 Bq/l.

Table 6 presents the measured values of gaseous and liquid releases for 1993.

The maximum of the monthly average values of the gaseous emissions do not exceed 1.1% of the established secondary level limits.

The evaluation of the mean exposure of the population is done by mathematical-model methods using the actual data on the amount and the composition of the released aerosols and gases and the specific meteorological parameters. It does not exceed 0.06% from the background exposure in the radiation controlled area [20].

The recorded levels on the same region resulting from the liquid effluents radioactivity, are significantly lower than the background ones. This is due to the exclusively low limit for the specific radioactivity of the effluents, which is close to the one for drinking water. The mean population exposure in the 30 km area, due to the liquid effluents is in the range of (2.84E-4 ÷ 4.13xE-4 manSv/a) [13], as the permissible annual limit of human exposure is 4.6 manSv/a [21]. It is proved by the use of common patterns, that licensing the annual limits for radioactivity of 0.74 Tbq (excludes tritium) and of 185 Tbq of tritium, released from NPP 'Kozloduy' through liquid effluents, ensures no violations of the dose limits and is economically reasonable.

4. Radiation monitoring in case of accidents

4.1 Automatic radiation monitoring

The automated system 'Berthold' for continuous radiation monitoring of the environment operates at 10 control stations. Two of these stations, the basic ones are located at the site of NPP and the rest eight are in the 1.8 km area. Dose rate and I-131 ground level concentration are measured at both types of control stations.

The dose rate in each of the stations is measured by two detectors. For the base stations the measuring ranges are between 0.04 ÷ 5000 μSv/h and 1mSv/h ÷ 1000 Sv/h respectively.

The iodine monitors are provided with 3"x3" NaI(Tl) detectors, as for the base stations the air flow through the iodine cartridges is 5 m^3/h, and for the control stations it is 1 m^3/h. Radioiodine accumulation and its spectrometric measurement are carried out simultaneously. For the base stations cycle continuity is 24 h, and the minimum detectable activity is 0.9 Bq/m^3. For the 10 min cycle of the control stations MDA is 500 Bq/m^3.

4.2 Meteorological observations

Nearby control station 4 a meteorological station is built up, according to the requirements for a third class meteorological station, including actinometric measurements. The chosen meteorological site is a typical place for the 30 km control area. The meteorological-station is MC 10 (EKO INSTRUMENTS TRADING CO. - JAPAN) and the following measurements are carried out:

- wind velocity and direction at 10 m height
- ultraviolet radiation from 0 to 0.14 kW/m^2
- global solar radiation 0 to 0.14 kW/m^2
- net radiation from 0.2 to 1.2 kW/m^2.

Atmospheric stability classification index by Pasquille-Horner scale, on the basis of instrumentation measurements and the method of Dr Senshu [9] is determined at the meteorological station. The obtained data are transmitted automatically to the auxiliary control room of the automated environmental monitoring system and then they are displayed. The meteorological station provides the input data for the patterned calculations in the 30 km control area of NPP. The code ARCAP computes the committed effective dose equivalents for individuals and offers decisions for protective measures on the basis of the accepted intervention levels.

A second meteorological station was projected in 1994. It will be provided with 52.5 m tower and is designed to carry out measurements of the temperature, wind velocity and direction gradient.

According to the „Convention of Early Notification of Nuclear accidents" the states, which have signed the convention are obliged to present and release the information for a nuclear accident at expected transborder transfer. Meteorological data required in such a situation are provided by the Research Institute of Meteorology and Hydrology (RIMH) including:

- wind direction with accuracy of 10°,
- major transfer - direction in degrees,
- wind velocity in m/s,
- observation period,
- mixing layer height in m.

4.3 Mobile laboratory for radiation monitoring

It is designed to perform environmental measurements in the event of an accident at a nuclear facility. The system is computer controlled for speed and simplicity. It is ruggedised in order to operate on-board a vehicle for mobile measurements. This is supplied with automatic self-contained sampling devices for aerosols, iodine and rain water; a gamma-spectrometer with a NaI (Tl) detector; equipment for continuous and punctual measurements of the dose rate. The control software provides fast initialisation of the sampling devices, measurement of the dose rate when the vehicle is moving, automatic gamma-spectrum analysis, data filing and processing of the dose rate and gamma spectroscopy records of the collected samples. Data transfer is performed by means of a radio-network.

Radiation monitoring availabilities in the country rapidly increased during the last years. For example, MH has been already supplied with sufficient laboratory and portable gamma-spectrometry, radiometric and dosimetric devices. It is coming forth the mounting of 30 'intelligent' detectors in a network, which covers the whole country. They provide permanent measurement of the ambient gamma dose rate and notification of the central office in case of exceeding of the set up thresholds. All that, combined with the equipment of the Universities, other research institutes and Civil Defence, ensures obtaining of the required information in the event of an accident or emergency.

III. General conclusion

NPP 'Kozloduy' has got the necessary technical equipment, a well-grounded programme, trained and motivated personnel to conduct an effective environmental radiation monitoring, which allows detection of any change in the radiological conditions. Environment in the vicinity of NPP 'Kozloduy" has been subjected to detailed and systematic studies by departmental services, surveillance institutions and independent experts. The results of this investigation are reported to the senior authorities and to the public.

The general conclusion is, that following the 20 year operation of NPP „Kozloduy" there are no statistically significant radiation and radioactivity changes and trends in the environment.

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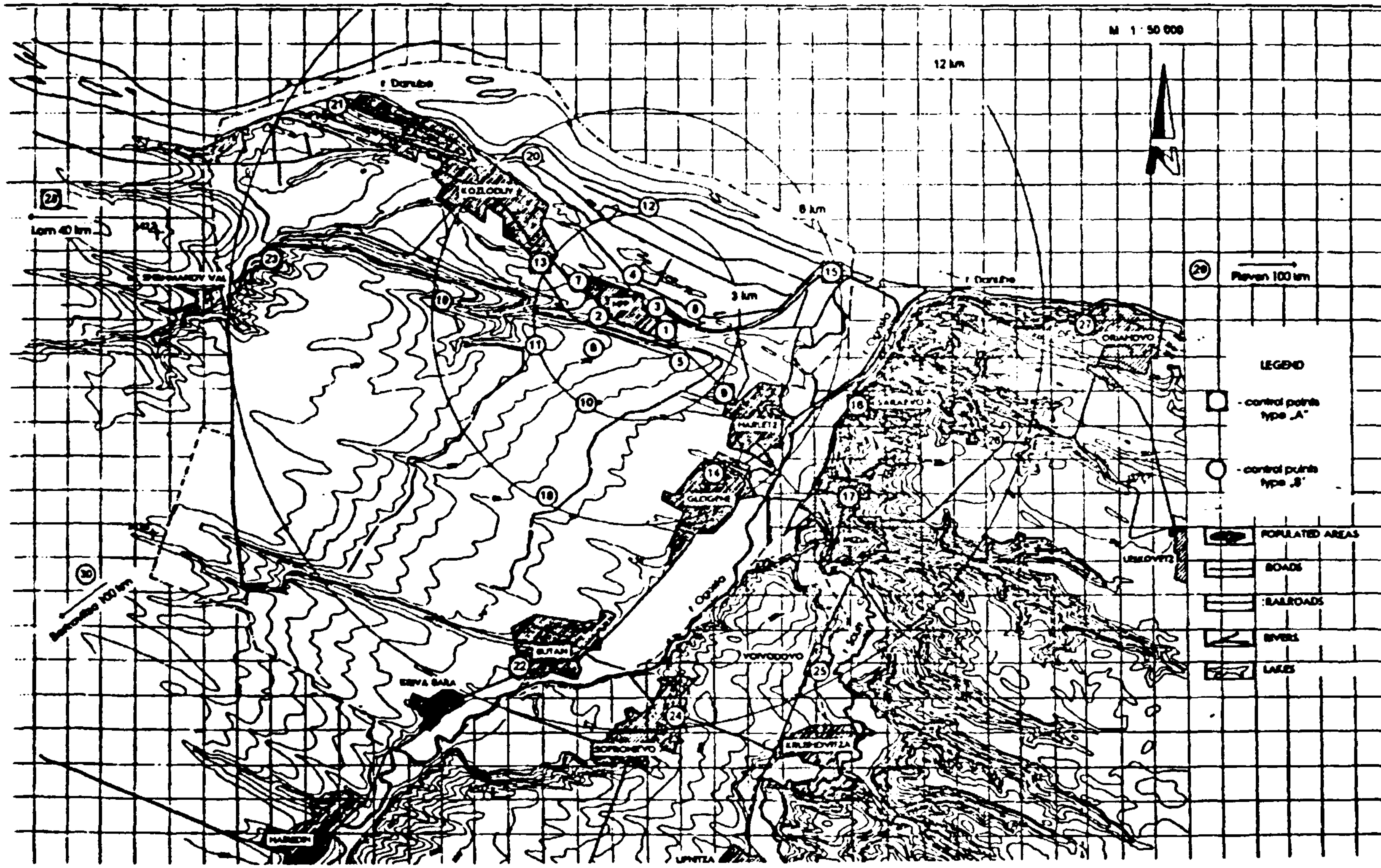
Figure 1

CONTROL POINTS LOCATIONS AROUND NPP "KOZLODUY"



SENIOR AUTHORITIES

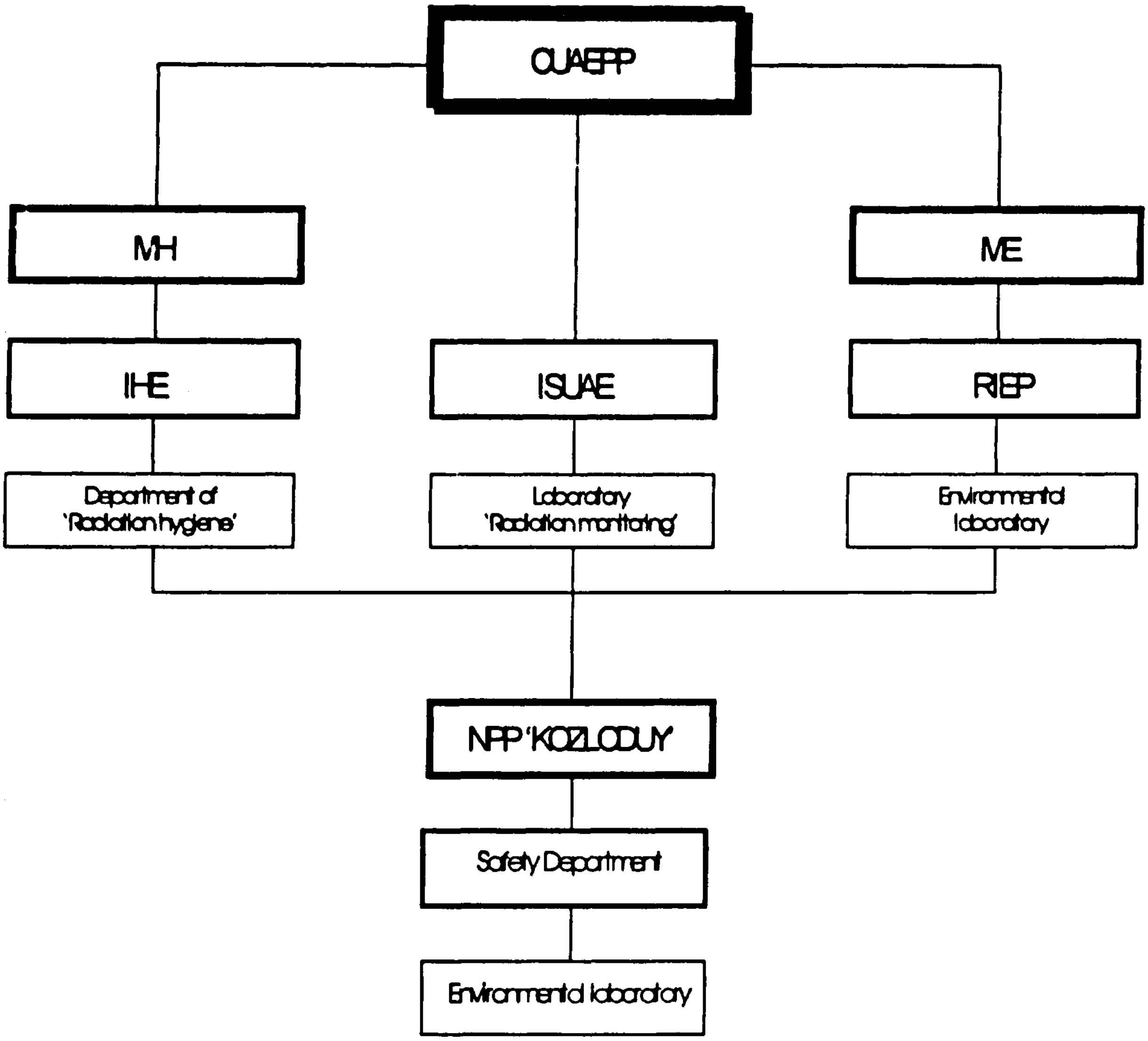


Table 1.

INSTRUMENTS USED FOR MEASUREMENTS OF RADIOACTIVITY IN ENVIRONMENTAL SAMPLES

Type of measurement	Instruments
1.Gross alpha/beta activity	Low background multiple detector , sample changer for low alpha/beta activities "Intertechnique"("Pegas") : - four gas flow proportional type counters - two guard counters - gas Argon - CH4 - automatic measurement cycles of 50 carriers (up to 200 samples) Low background beta counting system "Tesla" (type NZR-601): - one gas flow counter - one guard counter - gas Argon - Propan/Butan
2. Liquid scintilation spectrometry	Spectrometer "LKB" ("Rackbeta-215") Spectrometer "LKB" ("Rackbeta-1219")
3. Gamma - spectrometry	Gamma-spectrometer "Canberra", PCA-AT card "Accuspec", GeHP- detector, 20% relative efficiency, 1.9 keV resolution on Co-60 1332.5 keV line Gamma-spectrometer "Nucleus", PCA-AT card, GeHP - detector, 20% relative efficiency, 1.9 keV resolution on Co-60 1332.5 keV line
4. Dose measurements	TLD - LiF ("Alnor") Conventional survey meters with NaI(Tl) scintillators
5. Mobile laboratory	"Renault-Trafic" 4x4, gamma-spectrometer PCA-AT card, detector NaI(Tl) 3x3", 5 cm lead shield. Automatic sampling devices, 1 m^3/h, aerosol filter and iodine cartridge, rain sampling. Gamma detectors located on the side of the vehicle (10 nGy/h - 0.5 Gy/h), automatic data storage. Computer controlled system. Autonomy power supply.

Table 2

MINIMUM DETECTABLE ACTIVITY FOR ENVIRONMENTAL SAMPLE ANALYSIS *

Type of sample	Gross beta	Tritium	Strontium - 90	Caesium - 137 (gamma-spectrometry)
1. Water [Bq/l]				
1.1.Drinking water	0.1	6.9	0.001	0.006
1.2. Surface water	0.03	6.9	0.001	0.006
1.3.Undergroundwater	0.1	6.9	-	0.3
1.3.Water From inspection pits	-	-	-	0.3
1.4. Sewage and drainage water	-	6.9	0.001	0.02
2. Soil and bottom sediments [Bq/kg d.w.]	108	-	0.2	0.5
3. Algae [Bq/kg d.w.]	18.8	-	0.3	1.2
4. Grass [Bq/kg d.w.]	11.9	-	0.2	1.2
5. Food products (Corn) [Bq/kg d.w.]	4.7	-	0.07	1.2
6. Aerosols [mBq/m^3]	0.006	-	0.005	0.005
7. Air deposition (wet and dry) [Bq/m^2.day]	0.007	-	0.002	0.1
8. Milk [Bq/l]	0.8	-	0.005	0.3
9. Meat [Bq/kg]	1.1	-	0.02	0.4
10.Fish [Bq/kg]	1.9	-	0.02	0.4
11.Vegetables [Bq/kg]	-	-	-	1.2

* All MDA values are calculated in accrdance with the methodology developed by L.A. Currie, Analytical Chemistry 3(40)1968. Factors such as sample size, decay time, chemical yield and counting efficiency may vary for a given sample. These variations may change the nominal MDA value for a given sample.

Table 3.

Organisation of environmental monitoring of some NPPs in USA, Canada and Bulgaria [12]

Object	Number of measurement or sampling points					
	BNF	BNF	WBN	CANDU	Kozloduy	Notes
	1986	1991	1991	1990	1994	
Gamma radiation	35	41	43	25	46	
Aerosols	9	10	10	11	10 + 10 (1)	
Jodine-131	9	10	10	11	10	in air
Tritium	-	-	-	11	-	in air
Dry Deposition	9	10	10	-	36	
Wet Deposition	10	10	10	5	36 (2)	
Surface water	3	3	2	2	8	
Drinking water	4	4	4	4	4	
Underground water	2	2	2	2	182	
Bottom sediments	5	5	6	3	5 +12 (3)	
Soil	9	10	10	5	36	
Milk	3	3	4	4	3	
Fish	4	6	3	7	2	
Meat	2	2	2	5	2	
Grass	4	4	4	5	36	

(1) - The second 10 points are from the „Berthold"- system

(2) - The same points, as for dry deposition

(3) - 12 points from the drainage channels

Table 4.

Nominal values of MDA for environmental samples analysis of some NPPs in USA [15-19] and NPP „Kozloduy"

Nuclide	Water, Bq/l		Milk, Bq/l		Aerosols, mBq/m3		Soil, Bq/kg d.w.		Grass, Bq/kg d.w.	
	USA	BG	USA	BG	USA	BG	USA	BG	USA	BG
^{3}H	9.3	6.9	-	-	-	-	-	-	-	-
^{7}Be	1.7	2.7	1.7	2.7	0.7	0.04	3.7	3.9	18.5	11.4
^{54}Mn	0.2	0.3	0.2	0.3	0.2	0.005	0.4	0.4	1.9	1.3
^{60}Co	0.2	0.3	0.2	0.3	0.2	0.005	0.4	0.5	2.6	1.3
^{90}Sr	0.05	0.002	0.07	0.01	0.01	0.003	11.1	0.3	6.6	0.2
^{131}I	0.4	0.3	0.4	0.3	0.2	0.005	0.7	0.5	3.3	1.5
^{134}Cs	0.2	0.3	0.2	0.3	0.2	0.005	0.4	0.4	2.6	1.5
^{137}Cs	0.3	0.3	0.3	0.3	0.2	0.005	0.4	0.5	2.2	1.4

Table 5.

Some results of environmental monitoring of NPP „Kozloduj": preoperational period [6] and 20 years later

Object	Cesium-137		Strontium-90		Gross beta	
	1972-1974	1993	1972-1974	1993	1972-1974	1993
Water, Danube river mBq/l	4.0 ± 1.2	< 5.0 * < 5.0 **	12.0 ± 2.0	3.0 ± 0.3 * 4.0 ± 0.4 **	248 ± 70	220 ± 20 * 200 ± 20 **
Bottom sediments, Danube river Bq/kg d.w.	3.6 ± 1.4	27.0 ± 1.2 * 26.4 ± 1.2 **	2.6 ± 0.6	< 0.7 * 0.8 ± 0.2 **	889 ± 74	758 ± 81 * 500 ± 78 **
Soil, Bq/kg d.w.	7.6 ± 0.6	38.5 ± 22.8	5.0 ± 0.4	2.8 ± 1.5	703 ± 30	716 ± 64
Grass, Bq/kq d.w.	2.1 ± 0.1	< 3.2	4.4 ± 0.3	1.1 ± 0.4	963 ± 259	602 ± 289
Milk, Bq/l	0.13 ± 0.01	< 0.3	0.11 ± 0.015	0.02 ± 0.007	44 ± 1.5	40 ± 4
Meat, Bq/kg	0.30 ± 0.03	0.2 ± 0.08	0.09 ± 0.01	0.05 ± 0.03	67 ± 3.7	55 ± 6

* - Upstream ** - Downstream

Table 6.

Radioactivity of gaseous and liquid effluents from NPP „Kozloduy", 1993

Month 1993	Noble gases, Tbq	Aerosols ($T_{1/2}$/24h),GBq	Iodine-131, GBq	Liquid effluents, GBq
January	21.28	0.610	0.161	0.139
February	20.23	0.167	0.110	0.191
March	19.19	0.163	0.148	0.299
April	15.67	0.154	0.195	0.272
May	12.30	0.174	0.190	0.224
June	13.80	0.153	0.135	0.248
July	18.55	0.125	0.196	0.213
August	20.50	0.148	0.178	0.168
September	20.20	0.138	0.070	0.185
October	18.23	0.169	0.473	0.149
November	19.25	0.171	0.196	0.131
December	21.00	0.128	0.223	0.176
Total	220.20	2.300	2.275	2.395

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ENVIRONMENTAL RADIOACTIVITY OF THE DANUBE BASIN IN THE REGION OF NPP KOZLODUY (July 1994) *

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1. INTRODUCTION

The National Center of Radiobiology and Radiation Protection is carrying out regular radiation monitoring of environment in the region of NPP Kozloduy since 1972. As a hole, the results collected during the operational period (1974÷1993) do not contain evidence of any noticeable contamination of environmental materials caused by nuclear reactors of NPP.

In this paper the latest analytical results of July 1994 sampling for 13 sampling points are described. The location of sampling points and their distances from the NPP, environmental samples, analytical methods and radionuclides of interest are as follows:

Location	Water	Sediment	Soil	Grass
Novo Selo (r. Danube), 90 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Lom (r. Danube), 40 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Lom (r. Lom), 40 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
r.Tsibritsa, 20 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Kozloduy (r. Danube), 10 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Glojene (r. Ogosta), 6 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Butan, 10km			γ, Sr	β, γ, Sr
Orjahovo (r. Danube),13 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Krushovitsa, 14 km			γ, Sr	β, γ, Sr
Gigen (r. Iskar), 50 km	β, Sr, Cs	γ, Sr	γ, Sr	β, γ, Sr
Pelovo, 50 km			γ, Sr	β, γ, Sr
Cherven Brjag, 60 km			γ, Sr	β, γ, Sr
Dolna Mitropolija, 70 km			γ, Sr	β, γ, Sr

* Work supported by the National Electric Company - AD under the Contrakt No. 32/04.07.1994 - No. TC 9486009.

The meaning of symbols for analytical methods and radionuclides of interest is the following:

- β: determination of total β-activity;
- γ: determination of artificial γ-emitters concentration by high-resolution γ-ray spectrometry;
- Sr: determination of ^{90}Sr concentration by radiochemical separation;
- Cs: determination of radiocesium by radiochemical separation.

The location of sampling points is shown in Figure 1.

The main results of investigation are shown in Figures 2÷9. The data for the previous one-year period - May 1993 ÷April 1994 (*NCRRP. Radiation monitoring in the vicinity and in the region of NPP Kozloduy during 1993 and the evaluation of population doses. Final report for the Contract No.300116/31.05.1993, NPP Kozloduy, 1994* - in bulgarian) are also shown in this figures. Some results for pre-operational period, obtained in 1972÷1974 (*T. Petkov, R. Zlatanova, A. Kerteva, E. Novakova. Environmental radioactivity in the region of NPP Kozloduy in pre-operational period. Roentgenology and Radiobiology XV(1):51-54;1976* - in bulgarian) are used below too.

2. RESULTS

River water (Figures 2 and 3). Concentrations of ^{90}Sr and ^{137}Cs in the Danube water and in the water of inner rivers do not differ significantly. As in the previous one-year period, the values for ^{90}Sr are less than 10 mBq/l, and for ^{137}Cs are below the minimum detectable activity of the method used (2.4 mBq/l). The values are less than the corresponding results collected during pre-operational period (up to 34 mBq/l for ^{90}Sr and up to 22 mBq/l for ^{137}Cs).

River bottom sediment (Figures 4 and 5). Low concentrations of ^{90}Sr (less than 2 Bq/kg) agree with the tendency to decrease below those for pre-operational period (up to 5.5 Bq/kg). Cesium-137 content is varying in wide interval (6÷43 Bq/kg); nevertheless, higher values are in a range typical for environment contamination caused by the Chernobyl Accident.

Soil (Figures 6 and 7). As in the previous one-year period, values for ^{90}Sr are less than 4 Bq/kg. Cesium-137 content does not differ significantly from previous values also, and is varying in the wide range up to 80 Bq/kg. Relatively high values, measured in some sampling points in the vicinity of NPP (Kozloduy, Glojene and Orjahovo), are nevertheless in a range, typical for environment contamination as a result of the Chernobyl Accident. Such values were observed formerly even in the sampling points far from Kozloduy (Novo Selo, Pelovo and Dolna Mitropolija).

Grass (Figures 8 and 9). Values for ^{90}Sr in grass samples do not increase and are below 8 Bq/kg. Values for ^{137}Cs do not increase also, and are below 5 Bq/kg (and manyof them are less than minimum detectable activity of the method used, 3 Bq/kg).

3. CONCLUSION

The latest analytical results of July 1994 for 13 sampling points at distances 6÷90 km from NPP and the comparison with corresponding values for the previous one-year period allow to make the conclusion actual for April÷July 1994:

Radioactive effluents and atmospheric releases from NPP Kozloduy do not produce an noticeable changes of environmental radioactivity of the Danube basin caused by th global fallout and the Chernobyl Accident.

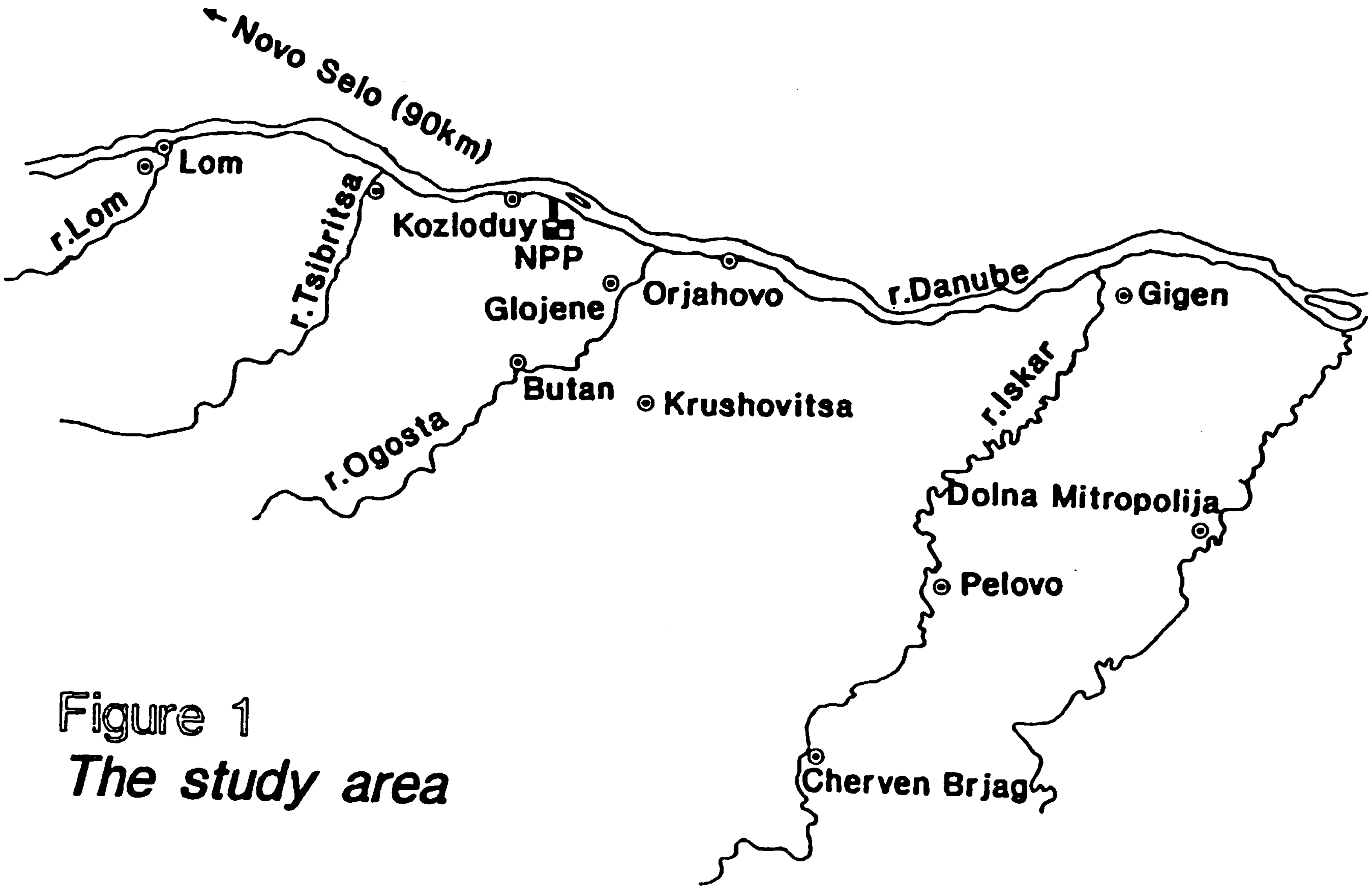


Figure 1
The study area

Figure 2
90-Sr in water, mBq/l

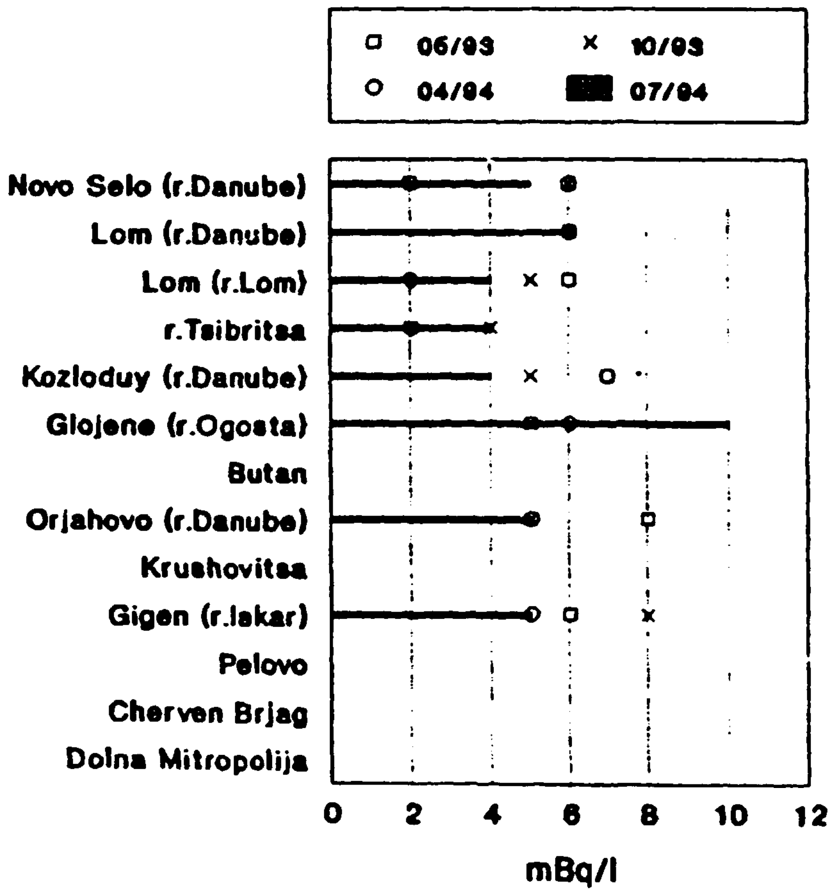


Figure 3
137-Cs in water, mBq/l

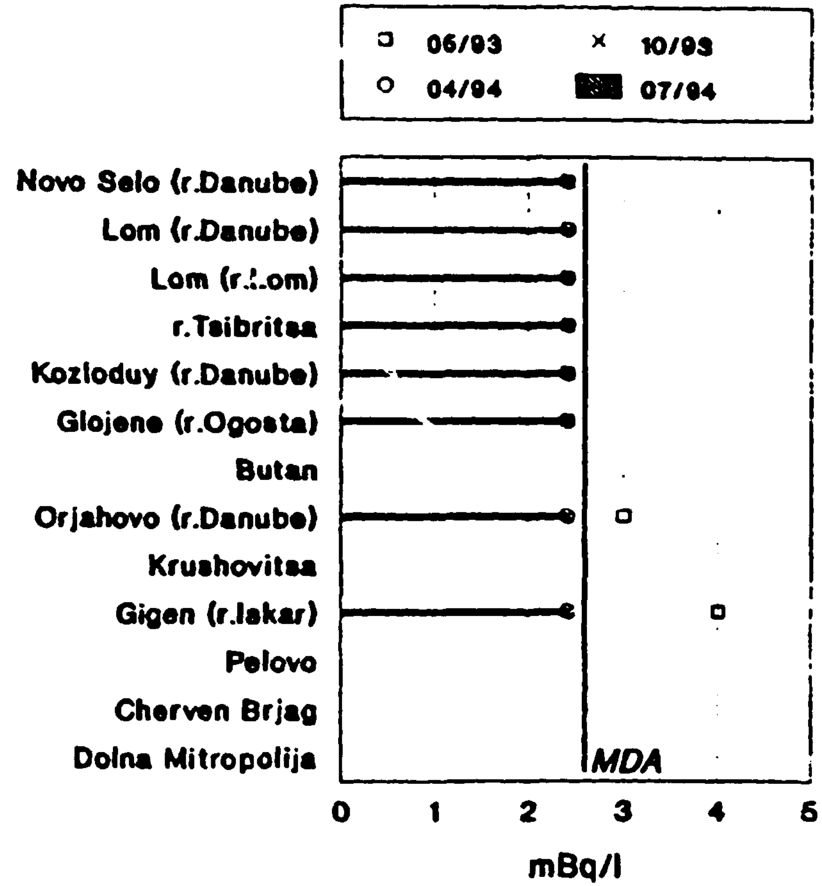


Figure 4
90-Sr in sediment, Bq/kg

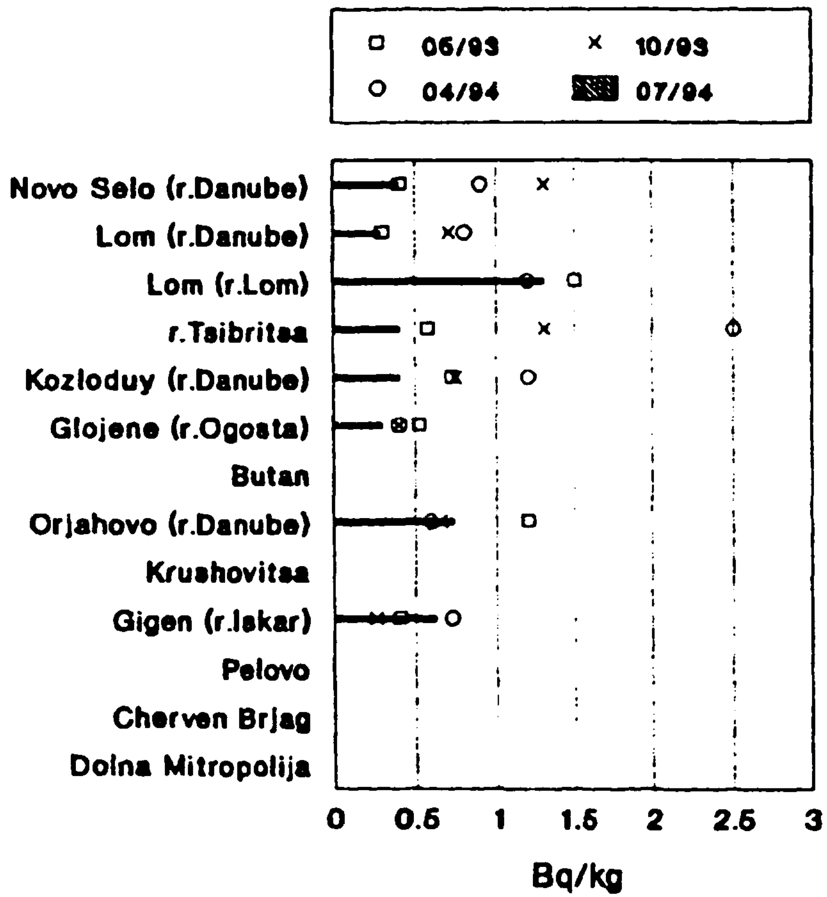


Figure 5
137-Cs in sediment, Bq/kg

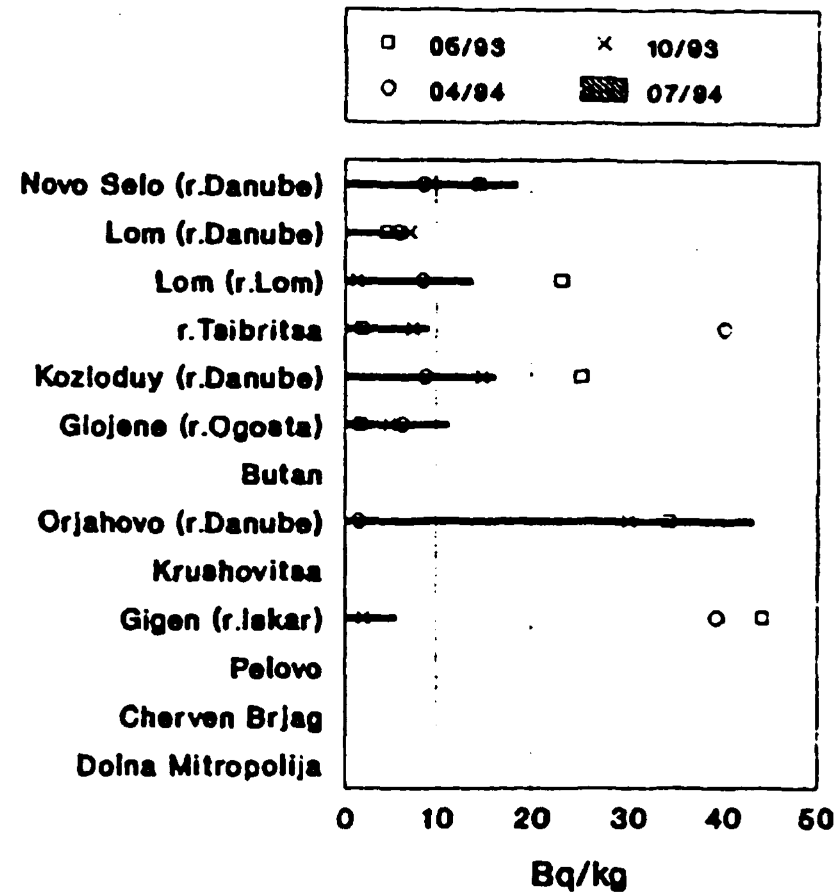


Figure 6

90-Sr in soil, Bq/kg

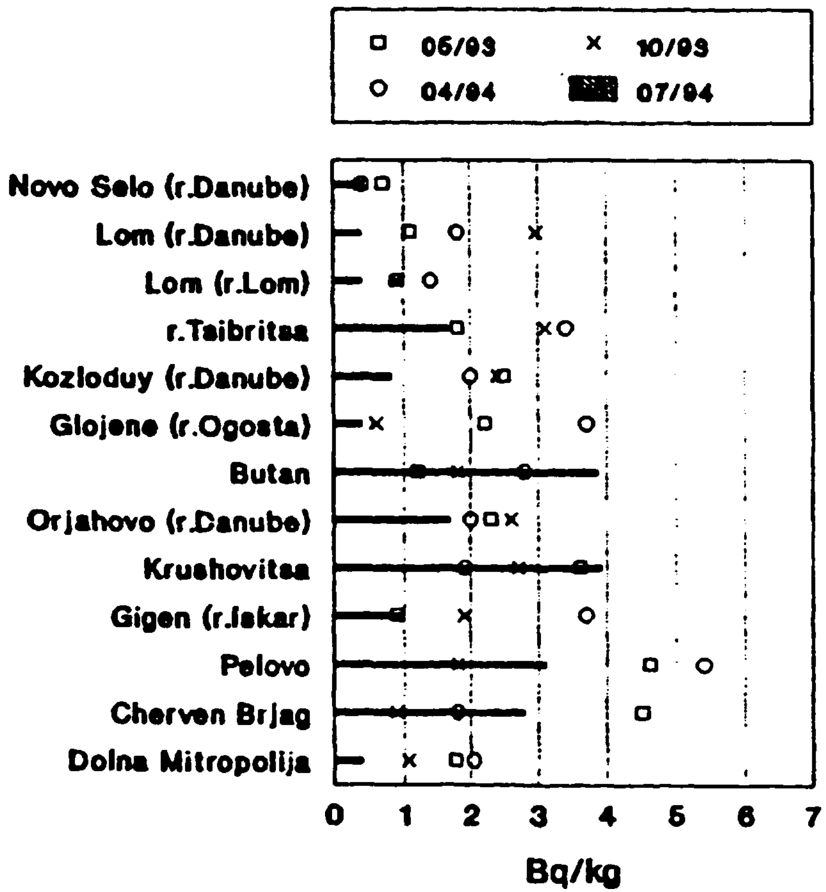


Figure 7

137-Cs in soil, Bq/kg

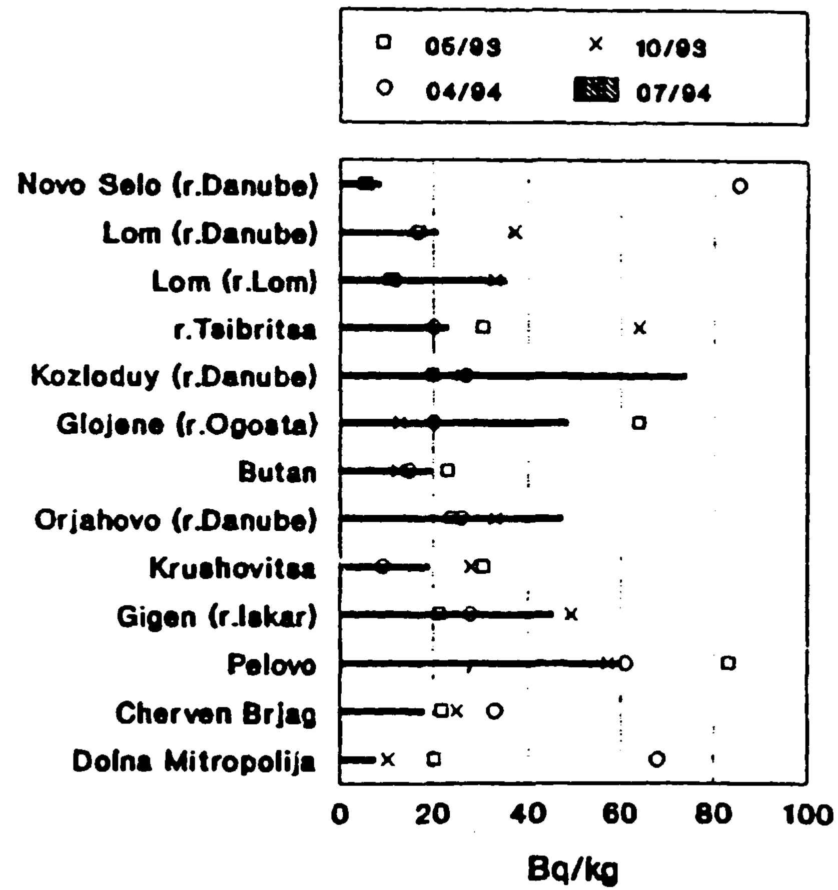


Figure 8
90-Sr in grass, Bq/kg

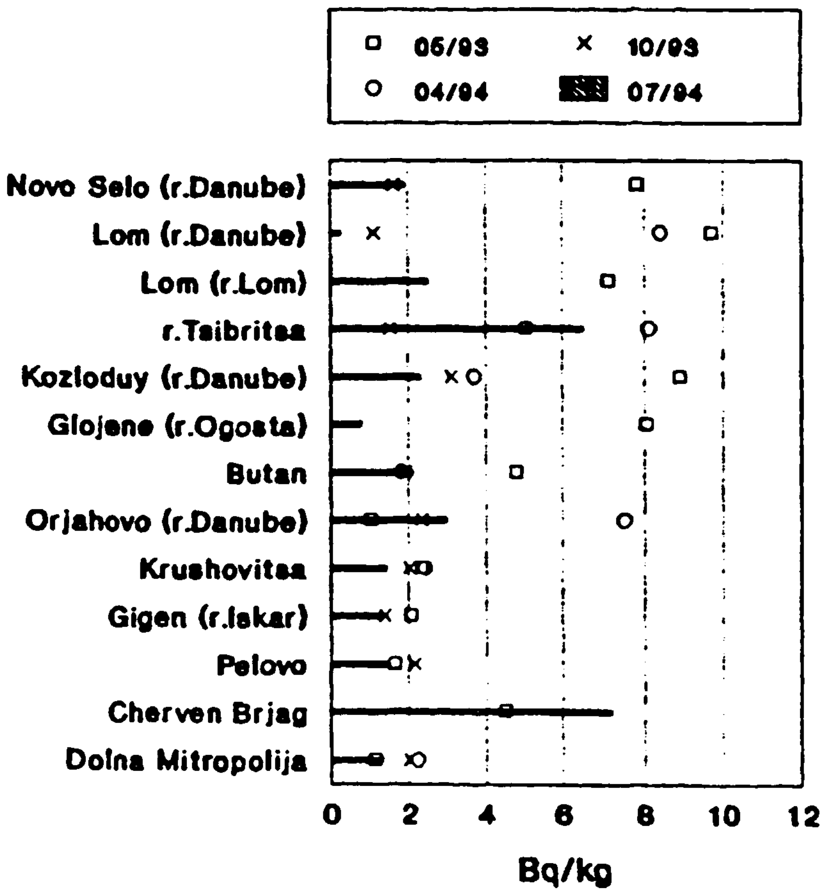
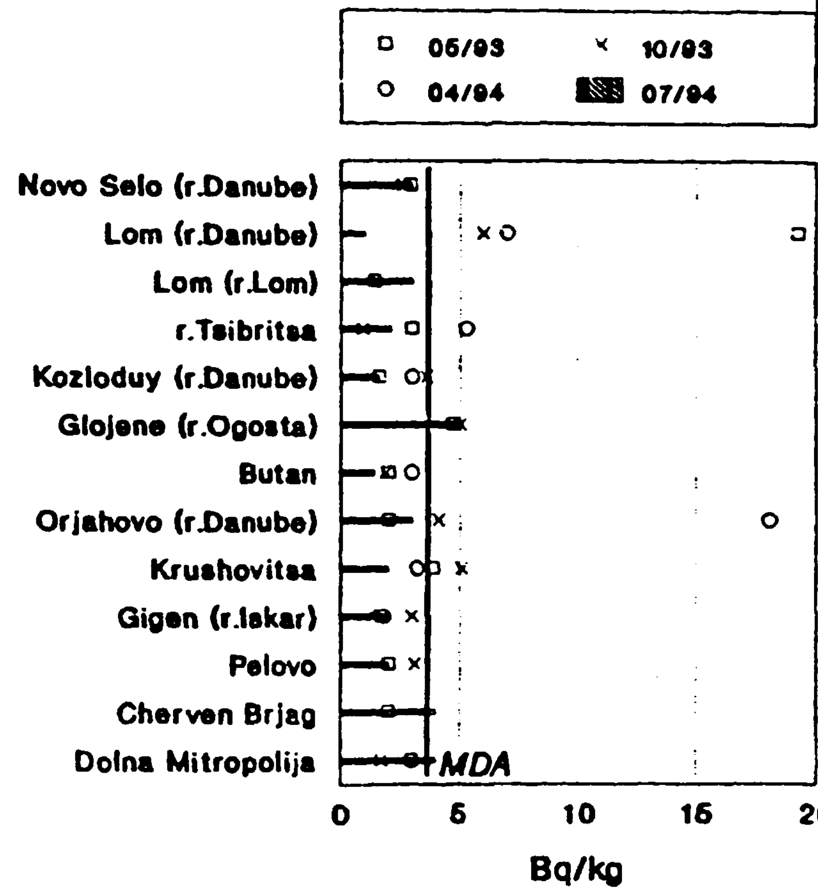


Figure 9
137-Cs in grass, Bq/kg



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ANALYSIS OF RESULTS FROM GAMMA-BACKGROUND MEASUREMENTS IN NORTH BULGARIA 30.5. - 3.6.1994

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The routine studies determining air dose rate from the natural radiation gamma-background and aiming at screening of technogenic radiation contaminations from the NPP "Kozloduy" led to the measurements made by three groups of specialists using four identical equipments in North Bulgaria residential areas, divided as follows: those belonging to the protective zone (3 km), to the examined zone (12 km), to the plume zone (80 km), as well as to control sights around the NPP.

The residential areas were divided between the groups: Group A - the first 10 residential areas (Table 1), Group B - from the 11th to the 26th, Group C - from the 27th to the 42nd. The points of measurement were chosen by local landmarks, far from buildings and over surfaces without additional coverings.

Another task was the determination of possibilities and sense of the applied methods of control on measurement precision and equipment stability.

1. METHOD:

Air dose rate from the radiation gamma-background was measured at 1 m above the earth surface, in the open, in accordance with the International Recommendations /1, 2/. The checking of precision (equipment stability) in three of the cases was made through comparison with control sources (radium 226) following each measurement, and in one of the cases - through the standard deviation in parallel series of measurements.

2. EQUIPMENT:

Four identical sets with similar technical parametres, including gamma analyzer connected to a scintillation probe V-A-S - 968 with a scintillator NaI (Tl). In one of the cases the results were printed automatically.

The instructions for the three groups aimed at criteria equalizing for fitting of the three equipments /3/. The fourth one was fitted twice, before and after the expedition, with the stationary equipment functioning at the Post for Non-stop Measurement of Radiation Gamma-background (PNMRG) at NCRRP. In this case the ratio $\frac{N}{N_b} = \frac{N_s}{N_{bs}}$ was valid. $N=N_{ss}+N_b$ is the sum of the counting rates when a control source is available and of the radiation gamma-background given by the portable equipment; and the values with "s" index are the same as the above mentioned ones but given by the stationary equipment.

This ratio was valid at s = 0.26 % because of the temporary fluctuations.

$\frac{N}{N_b}$ ratio contributes to the coefficient of correspondence between the calculated dose rate in air P_{SS}, made by the control source, and the determined dose rate from natural radiation gamma-background without a control source P_b:

$$P_b = \frac{P_{ss}}{\frac{N}{N_b} - 1}$$

During the second fitting of the portable with the stationary equipment another source was used. (!)

The duplicating equipment of Group C carried out parallel measurements at distance of about 1 m from the basic one. In this case no measurements with a control source were made. Gamma-dose rate in air was measured directly from the counting rate (the average one from the two ten-minutes measurements), taking into account a coefficient estimated in advance.

P_{ss} was estimated according to the formula: $P_{ss}=\Gamma_\gamma \frac{A}{r^2}$ [nGy/h], where Γ_γ - gamma constant of radium 226, A - the control source activity, r - its distance to the probe. Due to theoretical considerations P [nGy/h] = 8.67 P [μR/h].

3. RESULTS:

Tables 1, 2 and 3 sum up all the results. Determination of standard deviations are made separately for the results given by each equipment and group. Control sources are radium 226, four in number, with similar activity, with tolerance of 5%. The averaged results are from measurements with only one control source.

Results with numbers 27 - 42 are duplicated by the two equipments functioning simultaneously but with a different principle of control on their precision (results stability).

3.1. In case of minimum experimental error and stable equipment each row of experimental results $N_1, N_2, N_3, \ldots N_n$ and $N_{b1}, N_{b2}, N_{b3}, \ldots N_{bn}$ respectively, will have good precision, i.e. $N_{ss1}=N_{ss2}=N_{ss3}=\ldots=N_{ssn}=N_{ss}$; $\overline{N_{ss}} \rightarrow N_{ss}$ will be valid. $\overline{N_{ss}}$ - the average of the counting rates $N_{ss1}, N_{ss2}, N_{ss3}, \ldots N_{ssn}$ of the standard source. It is necessary to point out that there are two main reasons that make the latter condition not realizable in practice:

3.1.1. The counting rate of the background N_b, measured without a control source, cannot be determined simultaneously with the counting rate, which is theoretically identical but practically different, $N_b=N-N_{ss}$, participating in control source measurements.

3.1.2. Using portable equipment we have a strong influence of experimental errors in the statistics due to: differences in the geometry of measurements, uncontrolled nearness of the standard source, errors in measurements of mechanical values, imperfections in the control source, etc.

3.2. Control on precision through fitting with stationary equipment is also possible. The latter gives constant row of results with quite similar standard deviations by arbitrary but equal in their results groups. Variations in the standard deviation depend mainly on objective changes in the measured value and not so much on subjective ones, related to the state of equipment itself. At identical, fitted and in good shape equipments A and C, $\sigma_A=\sigma_C$ - up to the first and even to the second mark after the decimal point of the percentage /4/. Hence, when we have duplicated measurements by identical and in good shape equipments A and C, the condition for precision is $\overline{N_A}=\overline{N_C}$ and $\sigma_A=\sigma_C$ at simultaneous, as well as covering one and the same time measurements.

The limits for these conditions are criteria for results precision. The difference in average values gives the systematical error. Using two equipments with controlled and deliberately different spectral characteristics, an information on the qualitative differences in the nuclide structure of the sources can be obtained.

The fitting with a suitable base before the beginning and after the ending of the portable measurements is enough and it is an exceptionally exact criterion for the technical state of the equipment.

3.3. DISCUSSION:

3.3.1. Calibration under portable conditions leads to high standard deviation in the measurements of counting rates of the control source. (Tables 1 ÷ 3 - by Group A - σ_{Nss}=22.6%, by Group B - 8.2% respectively, and by Group C - 10.1%).

3.3.2. The fitting regarding the signal/noise ratio leads to additional systematical error (compare the results from the two equipments of Group C).

3.3.3. Calibration with different control sources leads to different results. This can be illustrated by comparing of the results from the simultaneous measurements with calibration of stationary equipment at PNMRG and the portable equipment of Group C, made before and after the expedition. The change of control source leads to systematical error of 24 %. At the same time the counting rate remains unchanged!Actually, the average value of the stationary equipment before and after the expedition is 8.1 μR/h or 70.2 nGy/h ± 2.4%, and for the portable one - 8.0 μR/h or 69.0 nGy/h ± 0.43 %, i.e. within the limits of the normal standard deviations values remain identical.

The parallel increasing of the average values and of the standard deviations is remarkable (Table 4).

Group	Average [μR/h] ± s [%]	σ_{Nss} [%]
A	11.2 ± 46.5	22.6
B	9.2 ± 23.3	8.2
	21.9 ± 19.0	10.1
C	7.7 ± 10.7	0.44

It can be expected that the increase in measurement precision will lead not only to decrease in standard deviations but also to decrease in average dose rates in air from the radiation gamma-background.

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Table 1. Results from the measurements in North Bulgaria 30.05. - 03.06.1994 г.

No	Residential area	N [imp./min]	N_b [imp./min]	N_{ss} [imp./min]	P_b rad. gamma-backgr. [nGy/h]	[μR/h]
1.	Malorad	2619	1481	1138	89.3	10.3
2.	Vratsa	p. 26			120.1	13.9
3.	Rogozen	6714	951	5763	55.4	6.4
4.	Burzina	9463	1527	7936	64.6	7.5
5.	Manastirishte	10118	1806	8312	72.9	8.4
6.	Vulchedrum	9133	1971	7162	92.3	10.7
7.	Lom	6215	2167	4018	179.7	20.7
8.	Montana	10113	1172	3941	43.7	5.1
9.	Mezdra	9855	2490	7365	113.4	13.1
10.	Botevgrad	12465	3894	8571	152.5	17.6
	Average ± σ [%] (without p.p. 1 & 2)	9260± ± 20.2 %	1997± ± 43%	7272± ± 22.6%	96.8± ± 46.8%	11.2± ± 46.5%

No	Residential area	N [imp./min]	N_b [imp./min]	N_{ss} [imp./min]	P_b rad. gamma-backgr. [nGy/h]	[μR/h]
11.	Mizia	11912	2402	9510	83.8	9.7
12.	Khurlets	13435	2016	11419	58.6	6.8
13.	Glozhene	12495	2010	10435	63.6	7.3
14.	Butan	11559	2084	9475	73.0	8.4
15.	Khairedin	12462	2272	10190	74.0	8.5
16.	Sofronievo	10721	1687	9034	62.0	7.2
17.	Saraevo	-	-	-	-	-
18.	Leskovets	12686	2855	9831	96.4	11.1
19.	Borovan	10983	1595	9388	56.4	6.5
20.	Banitsa	10022	1484	8538	57.7	6.7
21.	Lipnitsa	12240	2550	9690	87.3	10.1
22.	Voivodovo	13429	3055	10374	97.7	11.3
23.	Selanovtsi	13432	2970	10462	94.2	10.9
24.	Galiche	10120	1714	8406	67.6	7.8
25.	Krushovitsa	13448	3068	10380	98.1	11.3
26.	Vratsa	12142	3227	8915	120.1	13.9
	Average ± σ [%]	12072± ± 9.4 %	2333± ± 24.6%	9740± ± 8.2%	79.3± ±23.3%	9.2± ± 23.3%

Table 2. Results from the measurements in North Bulgaria - 30.05 - 03.06.1994 r. - continuation

No	Residential place	N [imp./min]	Nb [imp./min]	Nss [imp./min]	Pb Rad.gamma-backgr. [nGy/h]	[µR/h]	N [imp./min]	Nb ± σ [imp./10min]	Nss [imp./min]	Pb Rad.gamma-backgr. [nGy/h]	[µR/h]	Notice
		- first equipment -					- second equipment -					
	* Sofia - before the measurements	-	-	-	-	-	9773	*50471 ± 0.16	4725	+71.8/70.5 +68.7	+8.3/8.1 +7.9	+Basic equipment at PNMRG
27.	Altamir	18419	5480	12939	152.3	17.6	-	50003 ± 0.35	-	68.0	7.8	Under the fraction
28.	Turnava	22951	10223	12728	288.8	33.3	-	41650 ± 0.15	-	56.6	6.5	line - portable
29.	Biala Slatina	21330	6467	14863	156.4	18.0	-	53308 ± 0.43	-	72.5	8.4	equipment.
30.	Knezha	20450	7106	13344	191.5	22.0	-	59082 ± 0.04	-	80.4	9.3	The value below -
31.	Pelovo	21210	6681	14529	165.3	19.1	-	54920	-	74.7	8.6	by the counting rate
32.	Dolni Dubnik	18695	5970	12725	168.7	19.5	-	47931 ± 0.54	-	65.2	7.5	only.·
33.	Pleven	18259	5700	12559	163.2	18.8	-	45287 ± 1.26	-	61.6	7.1	* - Values from the
34.	Dolna Mitropolia	16659	6020	10639	203.4	23.5	-	53641 ± 0.1	-	73.0	8.4	portable equipment
35.	Trustenik	19480	6837	12643	194.4	22.4	-	42932	-	58.4	6.7	in Sofia, simulta -
36.	Krushovene	18351	7028	11323	223.2	25.7	-	55101 ± 1.0	-	74.9	8.6	neously measured
37.	Gulliantsi	16985	6404	10581	217.6	25.1	-	48102 ± 0.6	-	65.4	7.6	with the basic one
38.	Nikopol	17049	4975	12074	148.1	17.1	-	40089 ± 0.25	-	54.5	6.3	at PNMRG.
39.	Belene	17919	5706	12213	168.0	19.4	-	44664 ± 0.1	-	60.7	7.0	The measurements
40.	Svishtov	17844	5484	12360	159.5	18.4	-	47116 ± 0.46	-	64.1	7.4	in Sofia after the ex-
41.	Cherven Briag	17997	6589	11408	207.7	24.0	-	52611 ± 0.52	-	71.6	8.3	pedition are with
42.	Lukovit	19989	6208	13781	162.0	18.7	-	51806 ± 0.49	-	70.5	8.1	another st. source.
*	Sofia - after the measurements	-	-	-	-		7146 ± ± 0.39%	*50950 ± 0.68	2051	+56.3/53.8 +69.3	+8.4/8.2 +8.0	
	Average ± σ	18974± ± 9.1%	6279± ±18.4%	12566± ±10.1%	185.6± ± 19.1%	21.9± ± 19.0%	-	49265 ± 10.68% *50711 ± 0.47%	-	67.0± ± 10.7% +69.0± ± 0.43%	7.7± ± 10.7% +8.0± ± 0.44%	(by the counting rates only)

Table 3. Average values from those given in Table 2.

Group	N	N_b	N_{ss}	P_b	
	[imp./min] ±σ [%]	[imp./min]±σ [%]	[imp./min]±σ [%]	[nGy/h]± σ [%]	[µR/h]±σ [%]
A	9260 ± 20.2	1997 ± 43	7272 ± 22.6	96.8± 46.8	11.2 ± 46.5
B	12072 ± 9.4	2333 ± 24.6	9740 ± 8.2	79.3± 23.3	9.2 ± 23.3
C,1st equip. (results from the expedition)	18974 ± 9.1	6279 ± 18.4	12566 ± 10.1	185.6± 19.1	21.9 ± 19.0
C, 2nd equip. (results from the expedition)	-	4927 ± 10.68	-	67.0 ± 10.7	7.8 ± 10.7
C, 2nd equip. before the exsped. st.s.	-	5047 ± 0.16	-	70.5 ± 0.16	8.1 ± 0.16
C, 2nd equip. before the exped. st.s	-	-	-	53.8	6.2
c.r.	-	5095 ± 0.68	-	71.2 ± 0.68	8.2 ± 0.68
C, 2nd equip. after the exped. c.r. average	-	5071 ± 0.47	-	70.8 ± 0.47	8.1 ± 0.47
Basic equip - ment before, st.s.	-	-	-	71.8	8.3
Basis equip - ment before, c.r.	-	-	-	68.7	7.9
Basic equip - ment. after, st.s.	-	-	-	56.3	6.4
Basic equip - ment. after, c.r.	-	-	-	69.3	8.0

Notice: In the first column "c.r." - the result is calculated only by the counting rate at the proper coefficient;

"st.s." - the result is calculated by comparing with a standard source. The standard deviations in results from the basic equipment are usually under 0.5%.

GAMMA-RAY SPECTROSCOPY APPLICATIONS IN THE RADIATION CONTROL AND THE ENVIRONMENT MONITORING

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SUMMARY

Gamma-ray spectroscopy offers a rich information about the technological process at the NPP - Kozloduy and the radiation control on its territory as well as in the environment. Some possible ways of employing it, have been discussed in this report.

A method of gamma-ray spectrometers stabilization, which is especially needed for long-time measurements, has been considered. The possibility of efficiency autocalibration, when a refference source of the appropriate shape is not available, has been described. Some preliminary results about the effective thickness of extended sources assessment have been adduced, e.g. about the evaluation of the surface pollution infiltration depth, as well as about the development of pure beta sources (like Sr-90), using the registration of their bremsstrahlung.

ENERGY CALIBRATION STABILIZATION OF THE GAMMA-RAY SPECTROMETERS

The contemporary semiconductor detector gamma-ray spectrometers, providing high energy resolution, usually possess appropriate long time stability of the energy calibration. The energy calibration of the scintillation gamma-spectrometers is, however, not stabile enough. Despite their lower energy resolution as well, they have got a number of exploitation advantages - the comparatively low cost, the needlessness of liquid nitrogen and being of small size makes them preferrable to the semiconductor detectors in some cases.

A variety of methods and devices for the energy calibration stabilization are well known, described in the scientific literature or offered by series of firms. They are not only expensive, but when buying such systems, a preliminary particularization whether they would work steadily in the specific conditions, is indespensable. For example, the most often used method for locating the line, according to which the stabilization is done, is by comparing the counting rate in two single-channel analysers, put down on the slopes of a reference line. The ways of keeping the equality of the rates are different. The method works out satisfactory if the counting rate of the line is high and the line has a small background. When monitoring the radiation in the environment, both of the terms are not realized.

Proceeding from their own experience in scintillation gamma-ray spectroscopy and the fact that the scintillation detectors are now connected with personal computers, the authors propound an easy to accomplish stabilization of the registered gamma-ray spectra energy calibration. The amount of calculations and the memory needed are not too large and a personal computer af any kind, including a 8-bits, can be used.

Here is the scheme of this method application:

- the spectra are set up for a comparatively short time, assuming that the energy calibration would not change perceptibly;

- a reference line is chosen. The 1462 keV line of K-40 is propicios for the environment monitoring but if close lines (e.g. of Co-60) are expected to appear, the 2614 keV line of Bi-208 can be used. It, however, has a little more considerable statistics uncertainty;

- the position of the line is determined;

- the spectrum is recalibrated by the way of using numerical filtration or, if a programly regulated high voltage source is available, the needed correction is made.

Our experience shows that the most stabile and reliable results can be obtained if (after the background substraction), the position of the line is fixed by the approximation of its peak with the help of a second degree polynomial:

$$y = Ax^2 + Bx + C,$$

where y is the number of pulses in channel number x, and the coefficients A, B and C are determined by the least squares method.

Using the fact that the numbers of the channels are equidistant (contiguous whole numbers) and the co-ordinate system is so centered that the middle channel of the area we are interested in (containing n channels) is 0, the following comes out:

$$A = 15 \Sigma y (12x^2 - n^2 + 1) / [n(n^2 - n - 2)(n^2 - n + 2)]$$
$$B = 12 \Sigma xy / [n(n^2 - 1)]$$
$$C = 3 \Sigma y (20x^2 - 3n^2 + 7) / [4n(2 - n)(2 + n)],$$

which significantly reduces the amount of calculations, accelerates the procedure and minimizes the uncertainty evaluation. The value of x, for which the polynomial assumes its maximum, is taken as the line's position.

The numerical filtration with recalibration is done with the help of the following formula:

$$Y(z) = [\Sigma y(i + j) F(\delta i - j)] / a,$$

where Y(z) is the filtrated counting rate in channel number z from the new energy calibration (E = m.z); y(i) - in channel number i from the old calibration (E = k.i); δi - the fraction part from the calculation with whole-number values of z. A Gaussian with area 1 and width 1 ÷ 2 channels is taken as the filter core F; the summation is over j, which varies between ± (3 ÷ 4) filter widths; a = k / m.

EFFICIENCY AUTOCALIBRATION

The efficiency calibration is necessary in order to evaluate the activity of a nuclide in the measured sample by the area of the full-energy peak. The availability of a refference source with known activity, closest as possible to the measured sample, is requisite for the correct calibration implementation. The similarity of size and shape is most important for providing close counting geometries. However, some problems may arise, e.g. when evaluating the activity of an arbitrary piece of ore or metal, or in case the mass of the sample is little, but it can neither be taken as a point one nor fill the smallest available volume that we have got a calibration for.

The problem can be solved if there are nuclides with gamma-ray cascades in the sample. Let us take, for example, Co-60. Its both gamma lines with energies respectively 1173 and 1332 keV (and registered with areas S_1 and S_2) radiate contiguously in a very short interval of time and that is why a sum peak with energy 2505 keV and area S_{12}(reduced to a unit of time) appears. To a first approximation these areas are:

$$S_1 = A.\omega.e_1$$
$$S_2 = A.\omega.e_2$$
$$S_{12} = A.\omega^2.e_1.e_2,$$

where A is the source efficiency, ω - the solid angle, e_1 and e_2 - the photoefficiency of the detector for gamma-quanta with energies E_1 and E_2.

It is easy now to conclude that

$$A = S_1 . S_2 / S_{1\,2}$$

The consideration of the fact that some pulses from the photopeak coincide with pulses from the Compton distribution, leads to a more accurate formula:

$$A = N + S_1 . S_2 / S_{1\,2},$$

where N is the total counting rate in the whole spectrum.

It is obvious that the gamma-spectrum is quite adequate for evaluating the activity of the measured sample and it is not necessary to know either the gamma-spectrum calibration or the counting geometry.

The determination of N may cause some problems. Its contribution to the evaluation of the activity is usually about 20 ÷ 40% and if this precision is satisfactory, it can be ignored or a medium correction for the certain detector can be introduced.

A more precise result can be obtained if tabular or experimental data for the photopart of the detector is used.

The normalizing of the theoretical description of the Compton distribution to the Compton edge of the concrete spectrum seems the most accurate method. Using a program that realizes this procedure, in [6] we have clearly shown that for comparatively short measure times (200 ÷ 500 s), the activity of a Co-60-source, placed near the front of the detector, can be evaluated with total uncertainty less than 6 % (the total sum of uncertainties of categories A and B), if its activity does not exceed 40 kBq. When reducing the activity to 25 kBq, the total uncertainty becomes 1 ÷ 1,5 %.

THE EFFICIENCY THICKNESS (THE SELFABSORPTION) OF A SAMPLE EVALUATION

The area S_i of the full-energy peak number i of a radionuclide with activity A, when the radiation passes through an absorber with thickness d, is:

$$S_i = A . w . p_i . e_i . \exp\{-\mu_i . d\} , \qquad (1)$$

where ω is the solid angle, p_i - the yield per desintegration of tne gamma-rays radiating, e_i - the probability of its registrating in the full-energy peak (photoefficiency), μ_i - attenuation coefficient of the gamma-rays absorption in the concrete material.

The photoefficiency of the detector can be experimentally evaluated by an independant measuring; the values of p_i and μ_i can be found in the appropriate refference book; S_i can be experimentally measured. Consequently, two unknown quantities remain in equation (1) - the thickness of the absorber d and the product A.w. If two or more areas S_i of the same nuclide lines have been measured, after taking a logarithm of equation (1) and using the least squares method, it is easy to draw a system of linear equations for those unknown quantities. If the activity is distributed in the sample, we will evidently obtain only an approximate evaluation for the thickness of the layer containing the distributed nuclides, in this case (1) is not true. The accuracy of measurement can be evaluated, for example, with the help of the mathematical apparatus, described in [1].

We experimentally tested the method, using the nuclide of Na-22 with gamma-lines energies 511 keV and 1275 keV [2], the radiation of which passes through a lead plate with thickness 3.32 ± 0.01 g/cm^2, evaluated by a measurement. After 10 contiguous measurements, the thickness evaluation turned out to be 3.05 ± 0.07 g/cm^2 (the values of μ_i were taken from [3]). The same experimental values of the lines areas, but using the values of μ_i from [4], led to the result of 3.44 ± 0.16 g/cm^2. The pointed uncertainty is the mean square error of the measured values.

The measurement of the efficiency e_i and the thickness d were made with the same source with activity A_0= 11.8 kBq at the same distance (≈50 mm) and consequently the evaluation of the activity was between A = (0.989 ± 0.004). A_0 and A = (1.004 ± 0.004). A_0, depending on the values of the coefficients of the gamma-quanta absorption that were used. The measurements were accomplished with a scintillation detector with a NaI(Tl) crystal, its size being ∅ 50x50 mm, for 500 s measuring time.

Using of sum coincidences offers an interesting possibility of evaluating the absolute activity [5,6]. This way evaluated, the activity is A = 11.96 ± 0.44 kBq, the activity of the source being 11.8 ± 0.3 kBq. The implemented measurements prove that the application of these methods is easy to realize. Besides, the influence of the various sources of systematic errors is considerably reduced.

These two methods concurrent use will certainly be a useful complement to the variety of traditional methods of foodstuffs, water, etc. gamma-spectrometrical control, of the selfabsorption in the measured sample specification. Another, more interesting application, is its utilization for measuring of local radioactive pollution of the earth surface, concrete or asphalt coverings, etc., as these methods let us evaluate the pollution infiltration depth. The pointed differencies in the evaluations of thickness and the activity, depending on the tabular values of μ_i, can be used for specifying the latter.

Other parts of the gamma-spectrum can also give information about the thickness of the source. If there is an absorber between the source and the detector, some of the gamma-quanta will be scattered forward through small angles, which will reduce their energy a little and after their total absorption in the detector, they will be registered as pulses in the area between maximum of the backscatter and the full energy peak. This leads to two consequences:

- reducing of the measured spectrum photopart, which is easily noticed and can be quantitatively connected with the efficient thickness of the source, if it consists one nuclide only with a simple emission spectrum, e. g. Cs-137, Co-60;

- rising of the minimum between the Compton edge and the fullenergy peak line, compared to the maximum of the latter.

Some useful information can also be extracted from the analysis of the maximum which has been formed in the very beginning of the spectrum (between 100 and 200 keV, for instance), due to the repeatedly scattered gamma-quanta before they get to the detector.

DEVELOPMENT OF PURE BETA-SOURCES

Measuring or discovering pure beta-sources requires special methods of the sample treatment and their radiation registrating, which usually takes a long time. On the other hand, some of these nuclides, like Sr-90, belong to the biologically important components of the pollution that may be caused by NPP - Kozloduy, and express methods for their evaluation are absolutely needed.

A physical method upon which such methods can be based, is the bremsstrahlung of the beta-particles. It is continuous and, at least theoretically, extends to the maximum energy of the beta-spectrum. Sr-90, which has got two components with energy 546 and 2274 keV (the latter from Y-90), continuously distributes pulses, which do not form lines, and thus can be satisfactory registered when exercising gamma-ray spectroscopy over samples or the environment. For that purpose, when calibrating a gamma-ray spectrometer, its photopart should also be specified. It is desirable for the calibration to be accomplished with the help of refference sources, that contain one nuclide only. The nuclides that are expected to be found in the measured sample (often the measured nuclides, like Cs-137 and Co-60, have beta-emission besides, which is also registered) should be used. The total sum of pulses is also determined during spectrum-processing. The background and the areas of the registered lines, divided by the corresponding photopart, is substracted from this sum. If a statistically reliable count of pulses is left, it may be due to:

- weak gamma-lines that have not formed visible lines, or to secondary processes that were not accounted in the calibration course;

- the presence of a pure beta-radionuclide.

From a pessimistical point of view, to the first analysis of a radioactive pollution, the second possibility should be assumed.

We used this method when analysing the samples from the NPP - Kozloduy environment. In the specific conditions (a scintillation detector with NaI (Tl) crystal, ∅ 40x40 mm; a Pb-shield, 5 cm thick; measuring time 4000 s; the samples are put in typewriter ribbon boxes, ∅ 62x23 mm, volume 65 cm^3, mass between 60 and 120 g), MDA = 1000 Bq/kg was evaluated (MDA = 100 Bq/kg for Cs-137 and Co-60).

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THE RADIATION MONITORING AT "THE MARSH"

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This report represents a brief description of the results that were obtained during 1993-94, in realization of Contract No 1176 between NPP-Kozloduy and SRS-Sofia University "St. Kliment Ohridsky" [1,2]. In pusuance of the Law of Environment Protection [3], the NPP-Kozloduy financed the present work on purpose to obtain the necessary data:

- a complete exploration of the territories polluted;
- analysis of some variants of treating the earth masses, which would reduce their influence upon the environment and the population;

to get instructions from the special control departments and the municipality for the environment restoration, according to the legal documents, and providing:

- minimum expenses for the earth treatment and
- making the utmost use of the already existing pollutions.

SUMMARY

The data for the polluted with artificial radionuclides regions of the canal system (CS), EP-1 and EP-2, obtained until 1992, have been briefly discussed. The measurement methods that we used, have been described and their precision evaluated. The results from the regions survey and the agroecological characterization of the soils have been presented.

The following variants for the polluted earth have been discussed:

- burying the polluted earth ;
- chemical tilling and treating with zeoliths, or deriving the artificial radionuclides by an appropriate crop-rotation;
- creating an experimental field-range;
- creating an wood massive;
- no special treatment.

INTRODUCTION

For a long time (at least since 1979 [4,5]), NPP-Kozloduy has been throwing out water polluted with radionuclides, into the canal system that was costructed on purpose to dry up the territory situated between the NPP and the Danube, known as "the Marsh".

In soil-geographical respect, the Marsh belongs to the near - Danube soil region - a comaparatively narrow stripe of land, parallel to the Danube, including a variety of lowlands; the Kozloduy one, where the Marsh is placed, as well. The soil-forming materials in the real lowland, are carbonate alluvial depositions of the Danube and delluvial depositions on the slopes. In the higher region of the lowland, the soil-forming materials are represented by loess sediments. The underground water level in the different parts of the lowland varies, and is directly dependent on the Danube water level. There are conditions for marsh development in the low parts of the lowland, but most of them are dry now. Under the influence of a number of lay and hydrological conditions, in a temperate continental climate and meadow-steppe vegetation, the following types of soil have been formed: medow-alluvial and meadow alluvial/delluvial soils, carbonate black earth with no erosion and medow carbonate black earth with low erosion, typical black earth, and meadow black earth. All of them are agriculturally usable (to a certain degree) for a variety of cultures; appropriate agrotechnics (cultivation, fertilizing, watering, etc.) are naturally needed.

The high absorption of the loess that covers the territory, has prompted a natural cleaning of the thrown out water. The radioactivity has been concentrating in the bottom depositions in the canals. When cleaning them, the depositions were thrown out along the canals sides. This has led to the pollution of the territories laid along some of the canals. They are several metres wide with about 12 km total length. The most polluted regions, their area being about 400 daa, have been alienated and fenced in by wire-netting.

So it is important to accentuate that, for a nearly 20 years exploitation, this is the only significant radioactive pollution outside the NPP-Kozloduy territory.

There are some polluted spots in the very territory of the NPP. Some of them are on asphalt and because of their comparatively low radioactivity, can be liquidated by a new layer of asphalt. But the spilling of liquid radioactive wastes, most of all, has allowed the pollution of significant earth masses. Some of them are gathered in the so called "Gechova Mogila". The spots that are not liquidated yet, have been explored.

ANALYSIS OF THE DATA, OBTAINED UNTIL 1992

The fact that radioactive water flows into the CS of the NPP-Kozloduy, was registered for the first time in 1979 [5] and it has been the subject of many close explorations ever since then [6, 7]. The concrete sources, the mechanism and the ways of the CS radioactive pollution, have been described in some of them in detail [6].

The results from the aerogamma spectrometry research, implemented in 1990, are of an interest for the complete image of the pollutions in the CS region in NPP-Kozloduy. The conclusions are that the pollution caused by the nuclear weapons testing and the Chernobile failure, is comparatively low and homogeneous. Some radioactive pollutions of linear shape and directly connected with CS, have been located in the region, enclosed by the new canal, the NPP North fence, the working canal and the Danube. The seperate polluted zones are less than 20 - 30 m wide and several kilometres long.

Aerogamma spectroscopy data shows Co-60, Cs-137, Cs-134 in some regions of these anomalous zones, and just Cs-137 and Co-60 in others. The exposure rate in these anomalous stripes tends to a steady decrease the farther it gets from the NPP, but is rather irregular and ranges from several scores of μR/h to 200 - 300 and more μR/h (measured on the surface).

Later, in March 1992, an aerogamma picture of the CS region, was taken by Cousteau's staff as well. However, the results of it do not concretize the already known picture of the pollution.

The results, obtained by the NPP dosimetric control group are very interesting as well. They represent a gamma-survey of the CS [9,10]. The exploration offers a fine preliminary image of the basic pollutions surface distribution and significantly helps and supports further explorations. Relevant measurings have been made by a number of other authors, too [6, 11, 12, etc]. In fact, low activity has been detected in the surface waters in all of the monitorings, but there is high activity in the bottom depositions of the draining canals. Cs-134, Cs-137 and Co-60 are the basic artificial radionuclides that determine the bottom distributions activity. In 1992 the Sr-90 specific activity in the drainage bottom depositions did not exceed the limits, typical for the natural reservoirs.

An interesting result about the variation of the measured during 1987 - 1992 Cs-134/Cs-137 specific activity ratio, is shown in [12]. It is compared to the respective theoretical evaluation, had the flowing of radioactive water stopped. The difference between the curves points out to the fact that the polluted water continued to flow into the CS after that year as well.

The territory of NPP-Kozloduy has also been polluted with radionuclides (either fission or corrosion products) during its exploitation. It seems that there have been two ways of polluting:

- radioactive pollutions with polluted equipment and technological) mechanical carrying out;

- radioactive solutions splitting.

The radioactive pollutions mechanical carrying out has led to the so called "hot particles" spread [13, 14]. Despite their high specific activity, the hot particles cannot cause a serious pollution as they have a total activity because of their small size. That is why they have not been discussed in this report, but anyway, we have had in mind their presence as the getting of a hot particle into the sample can result in a significant error. The presence of a hot particle was detected at least once in the course of our measurements.

The most complete description of the polluted spots, existing in 1992, was made by Tsvetan Andreev.

METHODS USED

Gamma-ray spectroscopy

The samples measuring was done with the help of a NaI(Tl), ∅ 40 x 40 mm, scintillation detector, put in a lead shield, about 5 cm thick. The probe is connected with a 20 046 radiometer. The pulses get to a multi-channel analysator CANBERRA-30 and then to a 16-bites personal computer. For the spectra analysis, a programme that fits the Co-60 full-energy peaks (as a doublet) and the Cs-137 line (as a singlet). The specific activity of the measured sample is determined by the lines area through calibration coefficients.

The dry soil samples were grinded to powder and winnowed. The wet soils and silts were directly put in the measuring containers. So, the data for the specific activity relate to the samples in the same state they were taken. The samples were placed in plastic boxes (for typewriter ribbon) with size ∅ 62 x 23 mm and volume (65 $\pm$ 1) cm^3. The samples mass ranges between 60 and 120 g, dependent on their density.

The single sample measuring time was usually 4000 s. The background is about 17 imp/s in Sofia and about 12 imp/s in Kozloduy. Distinctive analysable Cs-137 and Co-60 lines, can be observed if the sample specific activity is more than 100 Bq/kg. This value is taken as the minimum detectable activity (MDA).

Besides the lines of full absorption areas, the total counting rate of every spectrum and the total gamma-activity are evaluated. This evaluation is not quite precise, of course, but useful for it offers information about:

- the gamma-activity that has not prompted measurable lines;
- the Sr-90 eventual presence. Its beta-radiation is registered by its bremsstrahlung.

The Sr-90 radiation registering efficiency was evaluated in most conservative conditions - 0,12 (imp/s) / (Bq/kg).

In 100 samples, in which spectra full absorption lines are not observed, the total counting rate exceeding the background is (0,93 $\pm$ 0,24) imp/s. Even if we ignore the K-40 and the U and Th contribution, we can consider the maximum possible Sr-90 (or other pure beta-radionuclides with similar maximum energy) specific activity, about 10 % of the limit (which defines the sample as a radioactive waste).

The average counting rate, mentioned above, relates to (260 $\pm$ 80) Bq/kg average specific gamma-activity, which is typical for most of the measured samples in which full absorption lines are not observed. This confirms the statement that the Sr-90 presence in the region explored, is not quite probable.

Typical samples from different places were also measured with the help of high resolution gamma-ray spectrometer (by ORTEC) with a GeHp semiconductor detector, with 24,9 % relative efficiency and FWHM about 1,8 keV.

The precise gamma-ray spectroscopy with semiconductor detectors enables not only the not detected by the scintillation spectrometry radionuclides measuring, but the radioactive pollution "age" evaluating as well (by using the difference between the Cs-134 and Cs-137 half-life).

The Sr-90 analysis was implemented with the help of Mayer and Scholl method which suggest Y drawing out of the sample. Sr-90 and Y-90 have to be in equilibrium for the samples, taken from the CS. The new thin sample is radiometred and the part of the registered pulses that is due to Y-90, is determined by the half-life. Thus the Sr-90 content is evaluated.

The Pu determination is done through a radiochemical seperation. The total alpha- and beta-activity measuring was completed with the French firm SHLUMBERGER's aparatus NUMELEC. It is 8-channeled and enables the simultaneous alpha- and beta-activity measuring.

TLD USAGE

The degree of the artificial pollution in the earth under the working CS, was of a great interest for the present investigation. Such detailed investigations are rather difficult for it is necessary to work under about 100 - 150 cm of water and from 5 to 30 cm of bottom depositions. Some attempts to take soil samples from the bottom of the canal, having already taken the mire away (with the help of an excavator), were made, but this procedure is technically hard to realize and disposes to rough errors. Here rises the necessity of finding new methods to solve the problem that are easy to realize and the probability of significant systematic uncertainties, is reduced to minimum.

A possible solution of the problem is the thermoluminiscence detectors (TLD) usage. They are put in a hermetic metal tube, which is vertically driven in the earth (in the bottom of the canal) to the depth wanted. TLD, designed out at INRNE and manifactured by the Bulgarian firm "Protexy CD ", were used. The detectors represent small cylinders with ∅ 5 x 1 mm, with thermoluminiscence material CaSO4:Dy.

The detectors (102 pieces) were fixed (a pair of two at intervals of 2 cm) upon Al foil and hermetized with polythene stripes. The new "cylindrical detector body", 102 cm long, was put in an iron tube (drill), 250 cm long. The tube was driven down in the place chosen (70 cm in depth, about 50 cm of which soil and 20 cm bottom deposition). The detectors were measured by the producer.

We consider the first results rather hopeful and interesting. A typical region (with about 170 - 190 mR exposition - that is the bottom depositions area), followed by sudden reduce to values, close to the background, can be seen. Of course, further explorations are needed for making more precise and certain evaluations of the method and the concrete results.

The precision of the final result - categorization by volumes with specific activity an a certain range - depends on a large number of both objective and subjective factors. Some of the evaluated final results uncertainty components, are: the typicalness of the sample; the way of preparing the sample for measuring; statistical uncertainty in the spectrum measuring; the mathematical elaborating of the spectrum uncertainty; the stability of the gamma-ray spectrometer; its efficiency calibration; the nuclides identification; the soil density; the homogenity of the pollution and hot particles presence.

The analysis of the results shows that the most serious problem, is the matchlessness of samples, taken from close places, due to the unhomogeneous distribution of the activity. That is why we think it reasonable to consider the final results within (0,5÷2) from the values given.

THE MEASURMENTS RESULTS

The polluted with artificial radionuclides earth in the NPP Kozloduy territory and the region of the CS was explored, according to the following scheme:

- marking the area;
- measuring the exposure rate on the polluted surface;
- taking samples from the surface and in depth;
- measuring all the samples with the help of a scintillation gamma-ray spectrometer;

- measuring a part of the samples with a high resolution GeHp gamma-ray spectrometer;

- analysis of some samples for detecting alpha- and beta-radioactivity, and evaluating the Pt and Sr-90;

- determinating the relation of the exposure rate to the specific activity;

- evaluating the volume of the earth polluted.

The exposure rate (ER) was measured at 4430 points, forming a net at intervals of 10 m along the canal and 2 m perpendicularly to the canal side (to the fence). The measurments were completed with SRP-68-01 unit at 10 cm from the surface. ER was measured with a SRP-68-03 unit with a hermetic probe at 250 points in depth along the canal sides, and at 110 points beneath the water, in the bottom depositions (the mire) surface, as well. Besides, we searched for pollutions in other regions, but no places where the activity exceeds the natural background, were found.

The relation between ER and the specific activity enables the usage of the amount of measured samples (from about 4700 different points), compared to the little number of the specific activity samples (about 300 pieces), for the more precise evaluation of the earth polluted volume and the artificial radionuclides total quantity. Unfortunately, this relation cannot be used in the measurements at EP-1 and EP-2 as the ER measurements are rather influenced by the radiating of the nearby standing buildings and radioactive solutions transporting pipelines.

The same specific activity of different radionuclides, prompts different ER. That is why we chose the following quantity K as a summarized value of the specific activities:

$$K = \Sigma A_i / MDK_i ,$$

where A_i is the measured specific activity of radionuclide number i, and MDK_i - the maximum permissible concentration (specific activity) of this radionuclide in the measured sample. This quantity is introduced like the effective or the equivalant dose [15] and has to be less than 1 so that the respective part of the earth polluted would not be considered a radioactive waste.

13 points were chosen; both ER (10 cm beneath the surface) and the specific activity of the respective samples taken from the surface, were measured in all of them. The relation between the two is approximated with a straight line:

$$ER = 23{,}7 + 47{,}9 \,.\, K \,.$$

where ER is in μR/h. This line's correlation coefficient is $r = 0{,}943$, which shows a comparatively good correlation.

Knowing the ratio of the Co-60 and Cs-137 quantities, and K for the region, we can evaluate the soil specific activity for Cs-137 and Co-60 separately.

As crirical influence levels, the 30 μR/h and 100 μR/h isolines can be chosen.

We suggest that no special work is done in the zones where ER is less than 30μR/h (which exceeds the natural gamma background in Sofia with 50%) and the specific activity is less than 13% of the treated as a radioactive waste.

When ER is 100 μR/h (which is one of the definitions of the radioactive wastes), the K evaluation exceeds the specific activity, determined by the kerma-constant, with 60%. At the same time the permissible specific activity for the Ra equivelant is about 2,5 times less. That is why we classify the earth polluted in two categories:

- ER between 30 μR/h and 100 μR/h;
- ER exceeding 100 μR/h.

SUMMARIZED DATA FOR THE EARTH POLLUTED

The summarized data for the earth polluted is shown in the following tables. The pollution depth is between 30 and 60 cm.

Table 1. Summarized data for polluted soil in the canal system

	ER [μR/h]	Area [m²]	Volume [m³]	A[Cs] [Bq/kg]	A[Co] [Bq/kg]	A[Cs] [MBq]	A[Co] [MBq]
Canal	30-100	30780	11241			16723	653
sides	>100	8340	4131			44892	10738
				Total pollution:		61615	11391
						1,7 Ci	0,31 Ci
				Total nuclides		- 2 Ci	
Bottom of MDC			13000	4600	1300	92000	8700
Bottom of canal D			6000	1300	1000	9400	7000

The data for the working at the moment canals bottom pollution (displayed in the last two lines of the table) should be considered just tentative; significant uncertainties are quite possible. Besides, one should have in mind that in many of the samples, taken from the bottom of the main drainage canal (MDC), there is muchmore Cs-134 (about 200 Bq/kg) points out to comparatively new pollutions, continuing nowadays may be.

The only spot that was discovered outside the CS fences, is clearly distincted on the aero-gamma spectroscopy picture, taken in 1990. The ER (measured from a helicopter) exceeds the background with 10 - 12 μR/h and seems a little higher than the ER that was measured around the canals. In fact the SRP-68-01 measurings, 10 cm beneath the surface, read 30 - 50 μR/h. The reason is that this spot is wider, and so is better distinguished for the detector that is in the helicopter, than the narrow stripes along the canals that are polluted. The spot is approximately round, about 50 m in diameter. In the surface layer to 10 cm depth, there are 1400 и 1700 Bq/kg Cs-137 (Co-60 was not located there); the lower layers are practically clean.

Only the regions on the EP-1 territory are shown in the table as there are no large undeveloped polluted areas on the EP-2 territory.

Table 2: Mean, total and sum activity A for the explorated regions on the EP-1 territory.

Nucl	Region No 1 Volume 9.5 m³ Mass 13.5 t			Region No 2 Volume 9,0 m³ Mass 12.8 t			Region No 3 Volume 5.5 m³ Mass 22 0 t			Region No 4 Volume 2.0 m³ Mass 3 t			Region No 5 Volume50.0+(25.0)m³ Mass 71.0+(33.5) t		
	Mean A [kBq/kg]	Total A [MBq]	Sum A [MBq]	Mean A [kBq/kg]	Total A [MBq]	Sum A [MBq]	Mean A [kBq/kg]	Total A [MBq]	Sum A [MBq]	Mean A [kBq/kg]	Total A [MBq]	Sum A [MBq]	Mean A [kBq/kg]	Total A [MBq]	Sum A [MBq]
Cs-137	37.8	510	550	9.9	130	148	3.2	70	78	0.8	2	18	41.6	2080+(1480)	2100 + (1490)
Co-60	3.2	40		1.4	18		0.4	8		5.8	16		0.4	18+(9)	

SOME POSSIBLE WAYS OF TREATING THE EARTH POLLUTED

The following variants of treating the earth polluted with artificial radionuclides in the CS region, have been discussed: - mechanical deriving of the artificial radionuclides;

- deep ploughing;
- chemical treatment (with zeoliths);
- burying of the earth polluted;
- creating of an experimental field - range;
- creating of an wood massif;
- no special treatment.

The mechanical desactivation can be done by taking away the surface layer of the earth, 20 - 50 cm thick in the concrete case (as deep as the pollution has been distributed), its area being about 8000 m², its volume about 4000 m³ and 5000 t weight.

When applying deep ploughing (25 ÷ 60 cm in depth), the surface layer polluted gets under the range of the vegetation roots; the deeper and cleaner parts of the ground get on the surface.

The earth taken away is in fact a comparatively fertile deposition derived from the bottom of the canals. It can be spread (that is another possible way of treating it) in a thin layer (10 + 20 cm) on the nearby lands and then buried 40 ÷ 60 cm in depth (by deep ploughing). Thus they will get under the roots of the cultures. Having in mind the agrometeorological assessment of the soils in the region its fertile layer being comparatively thick, it can be expected that the fertility of the ground would not be troubled. Besides, the radioactivity placed below, will not enter the alimentary chain.

This solutions, however, has some significant shortcomings:

- it requires a large amount of digging and transport work;
- it will provoke a strong negative reaction among the local population and the whole society, although the norms would be strictly kept;
- that reaction would be much stronger in case of a future pollution, caused by the CS.

A variety of methods for the chemical treating of the earth polluted with artificial radionuclides, have been described in the specialized literature. They usually tend to reduce the radionuclides transmission into the plants, cultivated in such territories, and just ocasionally - to derive the radioactivity. Despite the plain results, other countries already aim at commercializing such methods (patenting, selling of techologies, etc.). Bying them does not guarantee their applicability for the conditions are remarkably different. That is why we should ourselves work out own (for the country) science - applied methods for the population defence in case of a serious radiation failure.

Another possibility is the earth "washing". Ordinary solvents (such as water) are not effective, besides they are rather dependent on the biochemical processes in the soil - water system. Effective washing can be achieved by concentrated nitric acid, but it destroys the soil and its fertility. That is why this technology can also be treated as a perspective and current for future researches one, but inexpedient in the concrete case. There is some limited information that some firms implement such washing with the help of a cyclone separator. Still this is a rather difficult to realize and little productive method and we consider its implementation in the CS region inexpedient as well.

The zeoliths are good for detaining the radionuclides, but are not well known as a method for their drawing out of the soil. That is why we cannot reccommend them as an adequate techology.

The artificial radionuclides drawing out by an appropriate crop-rotation is also often mentioned as a possible decontamination of the earth polluted. Yet there are no well known results, and we can conclude from the data available that it is not quite expedient and could draw out not more than a small percentage of the radioactivity per year. Obviously, this technology is rather dependent on the concrete conditions and it cannot be recommended at the present moment as necessary data for the country is not available.

If the idea of taking away the earth polluted is accepted, then we can suggest several ways of buryng or making it foolproof.

The radioactive wastes cementing is one of the efficient methods for their prolonged storage, without any danger for the environment and the population. Unfortunately, this is a rather expensive method and has not been generally applied in NPP-Kozloduy yet. Using the earth polluted as bolus material of the cement solution, is an efficient modification of this method. As far as we know, the techologies of the future radioactive wastes revision factory, make no provision for sands or other bolus material usage for the cement solution. Still we find it expedient to study the possibility of such treatment. Cementing the earth in the CS region is not indispensable because of its comparatively low specific activity and the availability of other ways of treating it. But this is the most expedient way of safe storage of the most polluted earth on the EP-1 and EP-2 territory. Besides, a sgnificant part of it is sands and could be easily made foolproof without increasing the cemented wastes volume.

Burying in the Dry canal

The Dry canal can be used as an already existing depository of low activity radioactive wastes. This canal has not been used as a drainage canal (for drying the Marsh) for at least 10 years now. That is why it may be assumed that it has not been necessary for the melioreative system and it can keep working without it. It satisfies all radiation security requirements:

- the canal is covered with concrete floors and is placed in a dense clay soil that impedes the radionuclides migration. The samples taken from the outter side of the concrete covering do not show detectable artificial radionuclides lines (MDA = 100 Bq/kg), while the inner depositions activity is several of kBq/kg);
- there are about 1000 m^3 depositions in it now, created in the course of using it as a part of the so called "old system" for throwing out the debalancing and waste waters, which are some of the most active in the region. It can take about 2000 m^3 more. Covering the wastes with a layer of clean soil (30 cm thick) to the top of the concrete floors, will require about 2000 m^3;
- a part of the canal is below the area's level (the top of the covering is 3 m in depth). These parts can be totally filled up with earth polluted; the protective layer of clean earth can be put over it. This provides about 1000 $\div$ 1500 m^3 additional volume.

That is, the Dry canal can provide volume enough for all of the CS polluted earth safe burying. After covering it with 30 $\div$ 50 cm clean earth, the area should be afforested with appropriate trees and will not be of any danger to the population and the environment. This variant requires comparatively small expenses. An advantage of its is that the most polluted earth are already on its bottom.

The authors recommend the usage of the Dry canal as the CS polluted earth storage only as a reserve variant, in case the experimental field - range creating is not accepted. The transport expenses in the two variants, are equal, but the experimental field not only makes the earth polluted foolproof, but provides fine conditions for science - applied researches which are rather important for the country. The two variants concurrent applying is possible as well.

The authors do not recommend the usage of the Dry canal for burying the wastes from EP-1 and EP-2, as their specific activity is much higher. Neither the legal norms nor the practically necessary guarding of such a storage outside the NPP territory, are clear enough.

The present radioactive depositions in the CS region are of a comparatively large volume and low specific activity. That is why the building of a special storage that would be more safe than the Dry canal, is not needed.

CREATING OF AN EXPERIMENTAL FIELD - RANGE

The countries where nuclear weapons were tested, used the regions polluted with fussion products, for experimental ranges, where many studies have been done: for determinating the transfer factors "soil - plant - animal products (milk and meat)", the radionuclides absorption by the plants, the ways of its reducing and disactivation of the soil. Other countries, esp. those where nuclear power engineering has been developed, on purpose to preserve the environment and to defend the population against radioactive pollutions, have created artificial ranges. On their territory a variety of experiments have been completed (after polluting the earth with artificial radionuclides). Every country has its own experience and that is quite indespensible as the processes studied are much dependent on the soils, the vegetation cultures, the climate, etc. Besides, for a long time such information was considered secret.

In contrast to the reducing radioactive pollution, caused by the nuclear weapons testing in the atmosphere, the "contribution" of the nuclear complexes, including NPP-Kozloduy, is increasing. The international project, implemented about the Chernobile NPP failure, is rather indicative.

Bulgaria has never (to say nothing of its present economics state) been able to afford radioactively polluted fields, on which to study the ways of reducing the radionuclides in the plants cultivated (caused by the pollution of the soil).

It is necessary to make a decision about the future of the polluted regions around NPP-Kozloduy (polluted mainly with Cs-137 and Co-60). One of the reasonable, not requiring much expenses variant, is the creating of an experimental field - range on the territory polluted (or a part of it). This is a preferrable variant even if the expenses needed are commensurable with those needed for the classic burying of the earth polluted. That is why this unique chance must not be missed.

It is also important that experiments in real, not imitated conditions, in places close to those cultivated by the population (but on much more polluted ground), would reduce the social tension and the radiophobia, existing not only in the NPP-Kozloduy region. The annual publishing of the results would help the nuclear energetics prove that it does not contribute to the radioactive pollution in deserving alarm dimensions.

It is suggested to create experimental fields with size about 10 x 50 m (0,5 daa), covered with polluted earth, about 1 m thick. Having in mind the existing volume of earth polluted, it is suggested to create:

- 2 fields with maximum polluted earth from the bottom of the Dry canal. The preliminary evaluations show that it is about 1000 m^3, its average specific activity being 13 kBq/kg Cs-137 and 6,7 kBq/kg Co-60;
- 4 fields with moderately polluted earth (about 6 kBq/kg), of which there are about 3000 m^3 along the canals sides and probably 1000 m^3 more from the most polluted splits, which will be taken from the bottom of the canals.
- 2 fields with clean soil for control experiments.

The advantages of this variant are incontrovertible:

- a bias for implementing of science - applied experiments of an extraordinary importance for the country, is created;
- the utmost use of the already polluted with artificial radionuclides earth is made, and conditions in which some volumes that may be polluted in the future, are created, in a way that makes the environment absolutely safe and cannot prompt any objections in ecological respect;
- the territory suggested is one of the most polluted and is close to the other highly polluted regions, which will decrease both the transport expenses and the probability of radiational pollution during the transporting of the earth polluted.

The afforesting is an addition to the project for regulating the underground water level and draining the Kozloduy lowland, and the polluted earth in the CS region treatment. It provides:

- improving of the regional microclimate;
- using the trees in the series of actions aimed at the regions draining;
- bordering the polluted with artificial radionuclides areas and limiting the radioactivity distribution in the close regions (that may be caused by wind or water erosion);
- making use of territories for which there is no alternative solution;
- finally, wood material producing as well.

The typical conditions in the Marsh (in respect of the soil and esp. its hydrological specifics), form certain requirements for the sort of the trees. The poplar and the willow are determined by the many years afforesting experience along the Danube as the most congenial for the wood producing, protective functions, planting, etc., of all the trees and bushes that grow in similar conditions.

NO SPECIAL TREATMENT

Except the Dry canal, in which the specific activity of the bottom depositions is comparatively high (about 13 kBq/kg Cs-137 and 7 kBq/kg Co-60) that explains the high ER (reaching 3 mR/h), comparable to the permissible value for staff of A category premises only, the other parts of the CS region can be left in its present state. This can lead to breaking the legal norms only in a few small regions.

However, we do not find this solution satisfactory, for the following reasons:

- it can contribute to the radioactivity distribution in now clean territories;
- it requires constant control, maintaining of the fences and the marking of the region, which can finally lead to serious expenses;
- it derogates from the reputation of NPP-Kozloduy not only to ours, but to the world's society opinion;
- a large area is absolutely blocked for any other work in the bordered regions of the CS.

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BG9700048

THE ADVANTAGE OF SCANNING IN STUDYING TERRAIN CONTAMINATIONS

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Introduction

The task of studying terrains contaminated with radioactive substances could arise in different objects and circumstances. But not all of these contaminations are emergency and than the scanning will be a preventive one. Earnest radioactive accidents occur also frequently and do not concern only the Nuclear Power Plants. In this respect the NPPs have the advantage that there is a continuous control of qualified specialists and (beside the earnest nuclear accidents) the contamination on side the NPP usually is promptly recognized. As is well-known there are some cases where a grave radioactive accident arises in incredible circumstances with high-activity gamma-emitters and many days pass before its accidentally revealing contaminating meanwhile large terrains. Typical cases are those of Housten[1], Ciudad Juarez[2], Goiania[3], San Salvador accidents bringing along to human victims, many people irradiated above the limit and great material damages.

In the vast literature of radiological accidents with which we recently became familiar we did not discover the concept of „terrain scanning". In these accidents there always exists a panic and it is reasonable to seek firstly the places with high-activity sources or these contaminated with spilled radioactive substance resulting from unsealing. However in such a case there is a possibility to make many omissions and to waste much valuable time.

Not making an effort to include the different circumstances in which the scanning will give valuable information both in preventive and emergency terrain studies we shall make an effort to show its advantage in cases where this is possible.

Method of scanning

Scanning by helicopter of large terrains in relation with the gamma-radiation is well-known. This very expensive technique is created and developed because of the possibility to discover ores and its availability and readiness allows in some cases to be used to find out radioactive contaminations. This was done in Bulgaria after the Chernobyl NPP accident as well as for clarification of radioactive contamination distribution in the sewerage of NPP „Kozloduy". In both cases the working group of L. Kerbelov completed the studies using an unique 40 l NaI crystal property of the Geological Studies Administration. The equipment and the working group are available up to now and their inactivity is not justified excepting misunderstanding of authorities. The current state of our industry using strong radioactive sources creates a potential danger of their spilling and such cases are known. The regular scanning with this equipment of different country regions particularly ones representing potential radiological accident of this kind could save resources of our government if the accident could be recognized and localized in time. The above mentioned examples support this reasoning.

The gamma-radiation scanning by helicopter could lead to revelation of a radiological accident and its rough localizing. This must be followed by scanning giving the required „resolution" on the terrain surface coordinates.

In Fig. 1 to 3 we represent some examples of scanning by hand using gamma-radiometer. These results lead to the following conclusions in favour of scanning:

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- Fig. 1 shows a complicated picture of the gamma-field distribution along a sufficiently long region in which there are clearly distinguished peaks identifying the source of contamination;
- In Fig. 2 is shown an example of gamma-field topography. In this particular study the picture did not show some significant conclusions beside that very clearly outlines the way of spreading of radioactive contamination although with low activity
- The gamma-field topographic picture in Fig. 3 showed by two different scales is related to a field contaminaton with radionuclides. This picture is more informative than that in Fig. 2 as it shows a complicated structure of radioactive contaminations.

At least it is evident for us that the scanning in the light of the showed results gives very significant information than the simply „feeling" the terrain using radiometer. The one-dimensional picture given in Fig. 1 and the three-dimensional pictures given in the following figures show possibilities to find out peculiarities of the field which could forecast the reason and source of the contaminations. Especially a sufficiently well formed „hill" could indicate the availability of a high-activity source disposed under the terrain surface. At preventive studies the appearing of a reliable peculiarity of the field could give warning in time for the start of a coming strong contamination.

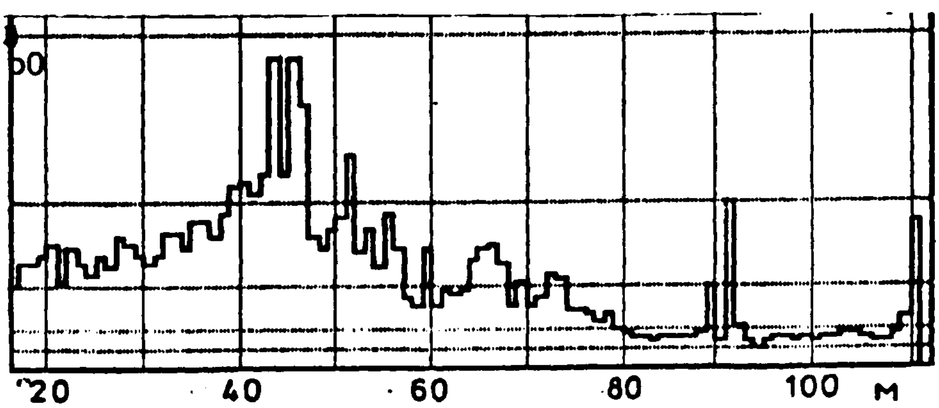

Fig. 1. Preventive linear scanning of a contaminated region

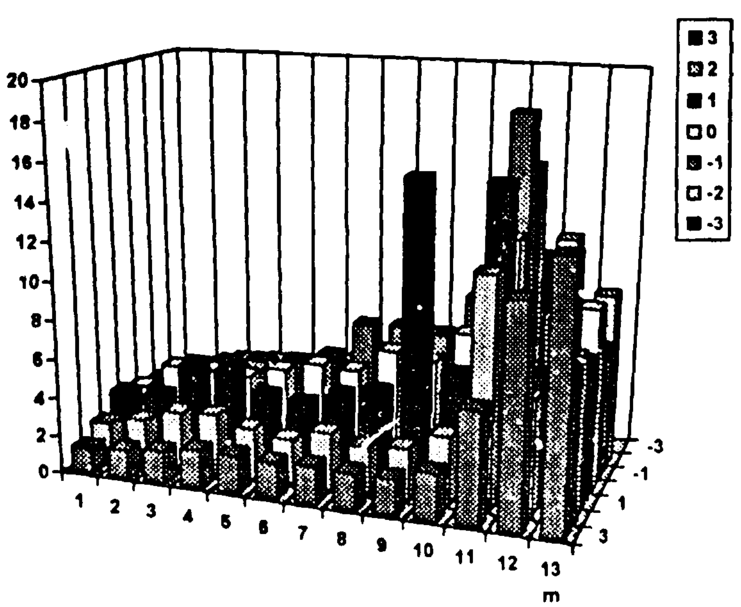

Fig. 2. Preventive scanning showing the gamma-field topological picture of a transport path region

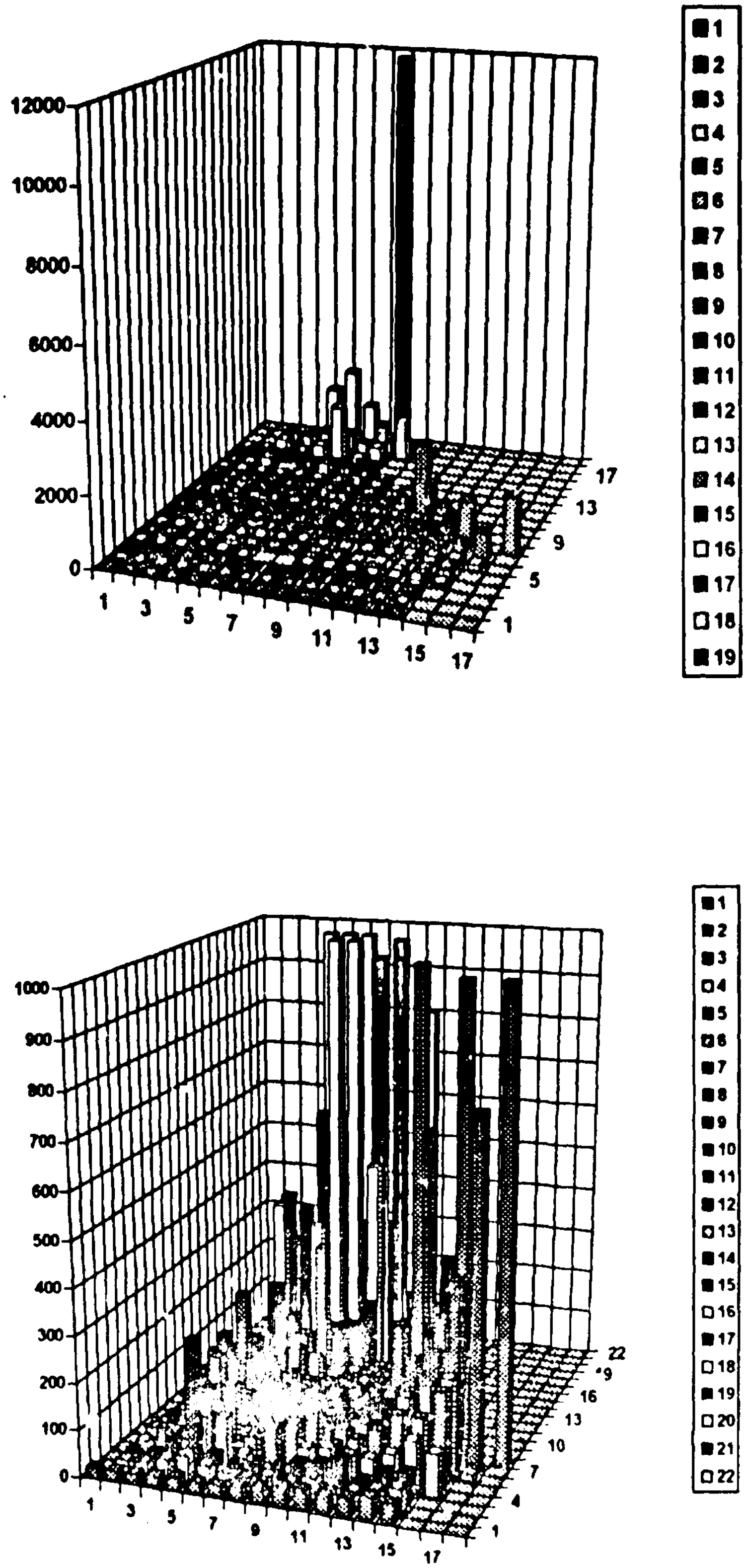

Fig. 3. Topological picture in two different scales of the gamma-field for a region contaminated with radioactive substances

Prospective

Scanning by hand of large regions is an exceptionally labour-consuming work. That's why it is necessary to construct systems for automatic scanning using detectors disposed on movable platform.

In cases where the scanning terrain has a complex profile the detectors have to be shifted at different heights in relation to the platform.

The scanning information would be more valuable if the gamma-spectrum is obtained by scintillation detector with NaI.

Using super-collimators (there is a separate report for them) would lead to much more precise localization of irradiation gamma-sources including those beneath the surface and with a complex topography.

In our opinion the means and time necessary to create a programme-supported scanning system would be many times justified in case of a grave radiological accident (not necessary in NPP) where the decision making would rely cnly to a real topological picture of the irradiation source distribution. Impossibility to obtain such a picture from the Chernobyl NPP accident has led to the irradiation above the limit to hundreds of people[5]. But the scanning systems will justify their creation as well as to the lighter cases even the preventive ones which was already mentioned.

Acknowledgment

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„SUPERCOLLIMATORS", ESSENCE AND SOME APPLICATIONS

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Introduction

Collimators of nuclear radiations are used in principle to form a narrow beam which directed to the central part of the detector leads to prevention of „edge effects" deteriorating the spectrum quality. By means of a collimator it is possible to determine sufficiently precisely the space angle under which the detector „sees" the irradiation source permitting to determine the absolute quantity of the activity. In Moessbauer effect studies the collimators decrease the „conic factor" which leads to decreasing the resonance line width.

In 1979 Paund and Wetterling proposed for the first time a special collimator type consisting of a capillary beam by means of which the resonance radiation of ^{57}Fe with an energy of 14.4 keV is „canalized" on the base of the total internal reflection[1]. Relying on the idea of these authors we decided to study the properties of such collimating systems conditionally refer to „supercollimators" for different radiations and aims. It turned out that the problem is more complex than it looks at first sight and here we shall present our first efforts in this field.

Essence and some main properties of the supercollimators

We shall name „supercollimator" a system of tightly arranged tubes whose length **L** is many times greater than their radius **r**. The space between the tubes is filled with substance for which we assume that (at the corresponding collimator length) it absorbs totally the radiation. That's why the radiation could pass only through the hollow of the capillary tube.

If the radiation source is placed tightly to one of the collimator walls than the intensity of the radiation passed through one capillary without accounting the scattering from the walls is determined by the space angle:

$$\Delta\Omega = 0.5\left[1 - \frac{L}{(L^2 + r^2)^{\frac{1}{2}}}\right].100 = 0.5\left[1 - \frac{1}{(1+\mathrm{tg}\alpha)^{\frac{1}{2}}}\right] \qquad (1)$$

where **r** is the capillary radius and **L** its length; the value gives the space angle part, 4π in %, in which the particles emit.

To illustrate the throughput of such a tube we shall point the following parameters: at r=0.3 mm and L=10 mm, i.e. at $\frac{L}{r}=33$, $\Delta\Omega=0.022\%$; at r=0.3 mm and L=100 mm, $\frac{L}{r}=333\%$, $\Delta\Omega=2.2\times10^{-4}\%$; at $\frac{L}{r}=1000$,

$\Delta\Omega=2.5\times10^{-5}\%$; at $\frac{L}{r}=10^4$, $\Delta\Omega=2.5\times10^{-7}\%$ and s.o. It is evident that to use really a supercollimator it must consist of great number of capillaries. Such beams are fabricated currently for microchannel plates[2]. The typical data of these plates are 10^5 channels with a diameter of 10 μm on 1 cm^2. It is easy to calculate that from an alpha-source stuck to such a collimator particles radiated from 30% of the surface will fall into the capillary. If the capillary beam has a length of 0.5 cm the space angle of every capillary would be

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$\Delta\Omega=10^{-4}\%$. In the case of plates with surface of 1 cm^2 on which there are 10^5 capillaries through the system will pass $0.3\times10^{-6}=0.03$, or 3%, of the alpha-particles radiated at unity time. This number is enough to use the supercollimator with this data.

An example of such a collimator usage is given in Fig. 1.

Evaluating the supercollimator properties the particles scattered by the walls and passed through the capillary must be taken into consideration. The scattering laws of alpha-, beta- and gamma-radiation especially at low angles are entirely different and depend on many factors: the energy, the disperser nature, the scattering angle[3,4]. As a result of scattering the monochromatic radiation spectrum becomes more complex and where the radiation is not monochromatic the picture is much more complex as is the case of beta-particles. Along with this in scattering at very small angle it is possible to expect radiation polarization. All of this makes the theoretical description of the radiations passed through supercollimators exceptionally complex and it is expected that the correct approach of the problem decision is the Monte-Karlo method. But in many cases this complex picture could help to study some „fine" interesting effects as is the case with the interference of Moessbauer and Rayleigh scatterings[5].

Currently we are far away of solving these problems but we submit some preliminary evaluations and the results of experiments like this in Fig. 1. Along with this we will show that before the supercollimator theory developing they could be used solving some practical tasks.

Alpha-radiation

The differential cross-section of alpha-particles elastic scattering at unity space angle is given by the well-known Rutherford equation:

$$\frac{d\sigma}{d\Omega}=\frac{1}{16}\cdot\left(\frac{z.Z.e^2}{E}\right)^2\cdot\frac{1}{\sin^4\frac{\theta}{2}} \qquad (2)$$

where **z** and **Z** are the alpha-particle and scattering nucleus charges, **e** - unit electric charge, **E** - the particle energy, θ - the scattering angle. $d\Omega=2\pi\sin\theta.d\theta$.

As a first approximation the alpha-particle scattering by the capillary walls could be described using the set up in Fig.2. The task is to find the scattering cross-section of particles scattered at minimum angle, θ_{min}, where they „slip" on the capillary surface to the maximum angle, θ_{max}, at which they still could come out of capillary.

$$\theta_{min}=\arccos\frac{x}{\sqrt{x^2+r^2}},\qquad \theta_{max}=\arccos\frac{L-x}{\sqrt{(L-x)^2+4r^2}}+\theta_{min} \qquad (3)$$

It is necessary to integrate at angle θ in the limits of θ_{min} to θ_{max}

$$\int_{\theta_{min}}^{\theta_{max}}\frac{\sin\theta}{\sin^4\frac{\theta}{2}}d\theta=-2\left(\mathrm{ctg}^2\tfrac{\theta_{max}}{2}-\mathrm{ctg}^2\tfrac{\theta_{min}}{2}\right) \qquad (4)$$

in the region of **x** to **x+Δx** and then the two **ctg** values in (3) are:

$$\mathrm{ctg}^2\tfrac{\theta_{min}}{2}=\frac{\left(\sqrt{x^2+r^2}+x\right)^2}{r^2}$$

$$\mathrm{ctg}^2\tfrac{\theta_{max}}{2}=\frac{\left(\sqrt{x^2+r^2}\sqrt{(L-x)^2+4r^2}+x(L-x)-2r^2\right)^2}{r^2(x+L)^2} \qquad (5)$$

Ultimately for the final macroscopic scattering cross-section along the capillary the integral is:

$$\sum(\theta) = n.k.\int_0^L (\sqrt{x^2 + r^2} - x)(\mathrm{ctg}^2 \tfrac{\theta_{min}}{2} - \mathrm{ctg}^2 \tfrac{\theta_{max}}{2}) \frac{1 - \sqrt{0.5\left(1 + \frac{L-x}{\sqrt{(L-x)^2 + 4r^2}}\right)}}{x - \frac{x(L-x) - 2r^2}{\sqrt{(L-x)^2 + 4r^2}}} dx \qquad (6)$$

where **L** is the capillary length. In this equation **n** is the number of the scattering centers (nuclei) of one capillary wall layer with thickness **d** referring to an element with atomic weight **A** and density ρ (g/cm^3):

$$n = \frac{\rho.N_L}{A}.d.4\pi^2.r \qquad (7)$$

N_L - Avogadro-Loschmidt number and **k** is a constant in the equation (1):

$$k = \frac{1}{16}.\left(\frac{z.Z.e^2}{E}\right)^2$$

The values of Σ (the macroscopic scattering cross-section) have been obtained for a capillary of gold, z=79, A=197, ρ=12.3 g/cm^3, E=5 MeV, d=10^{-4} cm at capillary length L=10 cm and different radiuses **r**. The data are represented in the following table where the first column gives the particles passed directly through the capillary and the second one - the particles passed through the capillary after one scattering.

Table in which values for alpha-radiation passed directly through the capillary and passed after one scattering by the walls at different capillary radius, r, and a constant length L=10 cm are given

r (cm)	Directly passed particles	Scattering cross-section, Σ
0.02	9.99×10^{-7}	2.24×10^{-8}
0.04	8.99×10^{-6}	4.49×10^{-8}
0.06	8.99×10^{-6}	6.74×10^{-8}
0.08	1.59×10^{-5}	9.00×10^{-8}
0.10	2.49×10^{-5}	1.12×10^{-7}
0.12	3.59×10^{-5}	1.35×10^{-7}
0.14	4.89×10^{-5}	1.58×10^{-7}
0.16	6.89×10^{-5}	1.80×10^{-7}
0.18	8.09×10^{-5}	2.03×10^{-7}
0.20	9.99×10^{-5}	2.26×10^{-7}

From the table it follows:

- increasing the radius the number of particles passed through the capillary and scattered from the wall increase;
- despite the complex functional dependence of Σ(**r**) according to equation (6) after the integration the relation between Σ and **r** is practically linear (this could be seen after plotting);
- the part of the scattered radiation is negligible compared to that of the directly passed radiation.

The last sentence has to be cleared. The values of Σ in the table referred to a very thin scattering layer with thickness of 10^{-4} cm (approx. 1 μm). If the layer is 10 times thicker (10 μm) than so many times increases the probability of alpha-particles passing through the capillary after scattering by the walls. But this case is not real one as alpha-particles with energy of 5 MeV loss entirely their energy yet in the first part of their run (the real run in the substance will be much longer because of the cosine factor); the maximum run of alpha-particles with an energy of 5 MeV in gold is approximately 10 μm[6]. The result is that the impact of the capillary wall scattering is negligible on the spectrum of the alpha-radiation passing through the capillary. The precise determination of this impact requires calculations based on Monte-Karlo method.

Beta-radiation

Beta-particles scattering laws at small angles are substantially more complex than those of alpha-particles and as was already mentioned in this case there is a problem with the continuous spectrum which will be differently distorted at the different regions after the scattering. Compared with the case of alpha-radiation here will be a complication for the particles passing through supercollimators - the braking radiation generation. We shall restrict our study to an example of obtaining an autoradiogram of ^{147}Pm beta-radiation passed through a supercollimator (Fig. 3).

Gamma-radiation

The supercollimators could be very interesting determining the low-energy gamma- or roentgen radiation on the base of the idea of Paund and Wetterling which was mentioned earlier.

The total internal reflection angle of the roentgen radiation is determined by:

$$\theta_k = 0.112.\frac{z.\rho}{A.E_\gamma} \text{ (grad)} \tag{8}$$

where **z** an **A** are the substance atomic number and atomic weight, ρ - its density (g/cm^3) and E_γ - the radiation energy (in keV).

In the case of a complex structure of the reflector the sum of z_i and A_i in percent are used.

Equation (8) shows that the critical angle value depends strongly of radiation energy. That's why the „transportation" of this radiation will be as more probable as more low-energy is the radiation. The critical angles have very small values. Concerning the roentgen radiation of iron at an energy about 6 keV these values vary from 0.02° of beryllium to 0.152° of gold.

That means that the intensities of directly passed through the capillary radiation and this due to the total internal reflection could be commensurable if the angle in equation (2) is commensurable with the total internal reflection angle. An autoradiogram of ^{57}Co radiation passed through capillary with a radius of 0.35 mm and length of 100 mm corresponding to $\omega \approx 0.2°$ is shown in Fig.4. There is an clearly outlined outer circle in the autoradiogram that is due to the total internal reflection of the roentgen line with energy 6 keV and to a certain extent to the gamma-line with energy of 14.4 keV. Unfortunately we do not know the content of the capillary glass and could not give a quantitative assessment of the reflected radiation intensity compared to the passed radiation. In addition for the precise quantitative calculations it is required to assess the contribution of Rayleigh scattering whose cross-section is very large at small angles and low energies[3,4]. In any case this result is worth studying further as there is a possibility electromagnetic radiations at different energies to be separated in purely physical way.

Some possibilities for practical applications of supercollimators

Nevertheless that the investigation of systems referred to as „supercollimators" is at its start our opinion is that there are possibilities at present of their application in some practical problems. For example:

1. By means of a supercollimator of gamma-radiation (i.e. with lead „filling") it is possible to construct „gamma-telescopes" specifying the topology of radiation fields. This could be very important in many cases especially at grave radiological accidents.
2. The supercollimator of alpha-radiation allows the assessment of emitter content in depth of a layer (e.g. of surface contamination) by the line broadening of the alpha-spectrum which is not influenced by the space angle.
3. The autoradiogram of ^{147}Pm beta-emitter in Fig.3 shows that by means of a supercollimator it is possible to identify the radioactive source nonhomogeneity.
4. The beta-radiation supercollimator allows the precise determination of beta-spectrum maximum energy using the half-weakening method[3].
5. The application of a magnetic field in the vicinity of a supercollimator allows the beta-radiation separation from the low-energy roentgen radiation which is essential in surface contamination studies.
6. The combination of a supercollimator and a thermoluminescent detector could reveal principally new possibilities determining low-energy roentgen radiations.

Many of these supercollimator applications have been already experimented and the results will be objects of new publications.

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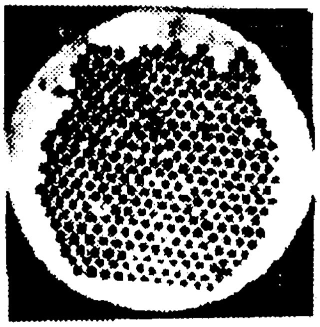

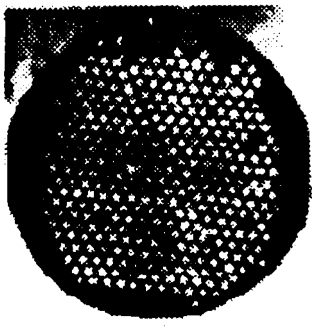

Fig. 1 Radiogram of ^{241}Am alpha-radiation passed through a beam of glass capillaries with a diameter approximately 0.7 mm and length of 10 mm. The alpha-radiation is developed on nitrocellulose solid-state detector (KODAK). The first image is obtained from a light beam passed through the nitrocellulose tape and the second - at crossed light polarizer and analyzer. This technique is described in[7].

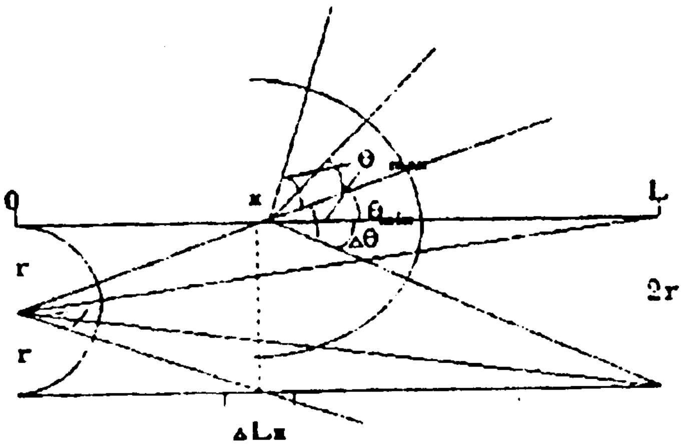

Fig. 2 Diagram of alpha-radiation scattering by capillary walls showing the meaning of the variables used in the equation

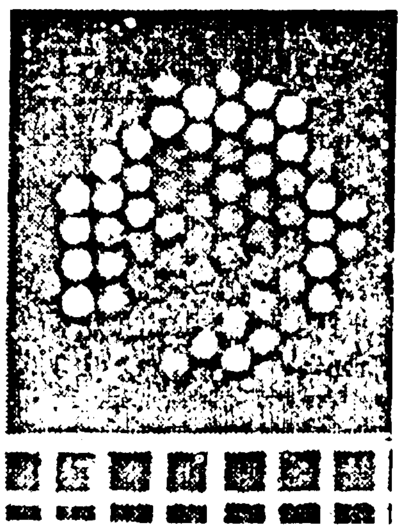

Fig. 3 Image on ordinary black-and-white film of ^{147}Pm beta-radiation passed through the capillary beam used for the radiogram in Fig. 1 (i.e. diameter approximately 0.7 mm and length 10 mm). Augmentation x11.

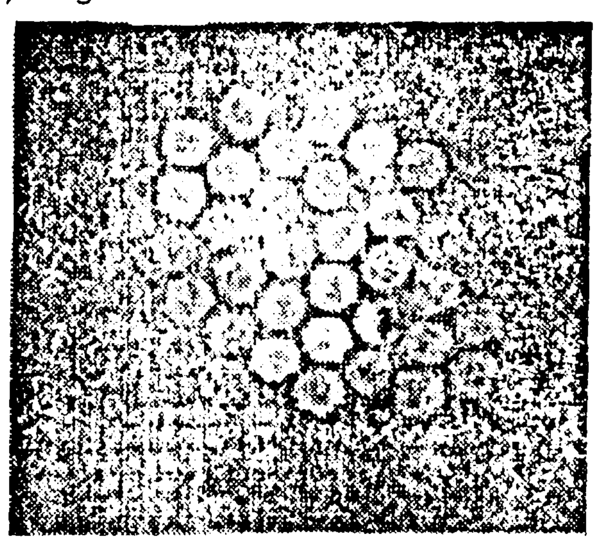

Fig. 4 Radiogram on ordinary black-and-white film for roentgen radiation with an energy approx. 6 keV and gamma-radiation with an energy approx. 14.4 keV of ^{57}Co through glass capillaries with diameter approx. 7 mm and length 30 mm. The film is in an envelope of black paper so that the circular images are not a result of glass fluorescence

BG9700050

ORGANIZATION AND TASKS CONCERNING POPULATION PROTECTION IN NUCLEAR ACCIDENT

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CIVIL PROTECTION OF THE REPUBLIC OF BULGARIA

Civil Protection of the Republic of Bulgaria represents an integral ingredient of the national security system. It features a compound of Government organizational, economic, scientific, technical and social activities aimed at the protection of the population in all emergencies. The Civil Protection (CP) infrastructure is based upon different Government acts and regulations.

The developments from global perspective and in the country at present determine the new tasks and goals confronting CP:

- Providing and creating the background for protection of the population and the property assets aimed at their survival in natural hazards, technological accidents, traffic incidents and warfare;
- Formation and training of forces and structures to exercise command and control functions related to civil protection activities, keeping high level of preparedness of these forces and institutions by dividing the country into four major zones of responsibility having relative autonomy;
- Education and training of the population on civil protection issues throughout the primary, secondary and higher school education system as well as in the professional CP qualification centers or by use of the mass media information system (radio and TV);
- Informing the population about the existing or potential risks of natural hazards, industrial accidents or traffic incidents.

The CP structuring is the responsibility of the national or local level government authorities, intended to provide for population and property protection in emergencies of all kind.

The basics of structuring and CP functioning can be formulated as follows:

- Reasonable sufficiency;
- Building up of the system based upon scientific analyses and concepts of the in-depth changes;
- Priority of the quality considerations;
- Focussing on peace-time tasks and missions complying with the international humanitarian law norms and principles;
- Dual contingency formation principle (professional and volunteer forces);
- Allocation of forces and assets according to the specifics and needs of the country regions.

The general command and control functions over CP activities are exercised by the Council of Ministers. The immediate management is handed over to the Minister of Defence, while the permanent management is entrusted to the Director of CP.

A Permanent Commission for Public Protection has been formed under the Council of Ministers. It exercises management, coordination and control functions during search and rescue operations, preventive activities to reduce and avoid the adverse effects of natural hazards, technological accidents and incidents.

The Commission is headed by a Vice-president of the Council of Ministers. Members of the Permanent Commission are ministers and heads of central authorities having responsibilities for the protection of the population.

In its extensive research and development work on civil defence projects and the process of taking scientifically based decisions the Permanent Commission is being helped by a Scientific Coordination Board with the participation of leading scientists and experts organized respectively in eight Expert Boards on nuclear and seismic safety, meteorological, hydrological, radiological, chemical, biological and medical protection.

The ministries, agencies, regional administrations, industrial establishments and companies have their own analogues of the Government Commission while the local administration CP staff officials have advisory (supervisor) and coordinating functions.

The restructuring of the CP system started in 1991. It was called to life by the vast changes in the country and worldwide. The CP functions and activities are by now much more in accordance with the international humanitarian legislation. At present CP represents an independent government institution of specific character. New organizational structure is being implemented. The focus of attention is placed on peacetime crises reduction and provision of professional assistance for the population and the national economy establishments during natural hazards and technology accidents of any kind.

The essentials of the conversion can be summarized as follows:

- Development of new legislation for public protection in the new realities. A new CP Law has been drawn up and presented to the Government awaiting approval by the Parliament;
- Active preventive work to reduce the adverse effects of hazards and accidents;
- Formation of professional contingent forces to handle emergency situations;
- Better public preparedness for protection and self protection;
- Establishing of closer bilateral and multilateral contacts for operative interaction with neighbor states and the international community on population protection issues.

The CP restructuring is carried out on the basis of a scientific concept about the new aims and goal for the time period up to 2000. The reform takes into consideration the lessons learned and the experience of the European states and other nations worldwide. The new conditions have been carefully analyzed with special attention to the most common potential risks posed in emergencies, including off-site release in case of nuclear power plant accidents. A special off-site emergency plan has been prepared in response to an accident at the Nuclear Power Plant. The Plan plays the role of a major instruction manual detailing the organizational. recovery, protection, radiological and chemical, medical-preventive and other activities related to protection of the NPP personnel, the resident population, property and cultural heritage assets on the NPP site, in the 30 km zone around it and the country territory.

A system for population protection in case of radiological emergency due to NPP accident connected with off-site or transboundary release has been arranged. Major objective of the system is the preparation of recommendations and implementation of activities for population protection in radiation emergency as well as for diminishing the effects of and recovery from nuclear accidents or radionuclide release.

A system for background radiation monitoring is put into action. Arrangements are made for installing a national network of intelligent detector devices located at vulnerable points where monitoring for transboundary release is vital.

A schedule for research and development initiatives has been approved by the President of the Permanent Commission. The expert boards on radiation safety and protection work on projects seeking practical solutions to issues closely related with nuclear and radiation safety. More important of these are:

- Analysis and assessment of the after-effects of predictable and unpredictale accidents at NPP;
- Computer simulation for the purpose of analysis of seismic stability, fireproofing and NPP equipment control;
- Organization of the national radiation monitoring and public protection system;
- Metrology of ionizing radiation, investigation methods and software aids;
- Protection and medical care aids for irradiation.

The CP suggested laboratory testing of a new herb extract containing biologically active substances showing decontaminating effect on human beings related to radioactive strontium, heavy metals, etc..

The herb extraction is perfectly suitable for employment in the food manufacturing industry as an ingredient to non-alcoholic and alcoholic drinks and a wide range of spices where a taste of bitterness is appropriate.

In the field of nuclear safety and radiation protection the Republic of Bulgaria follows strictly the principles of the International Law and the Conventions ratified after 1988 concerning :

- Assistance in case of nuclear accident or radiological events;

- Early notification on nuclear accident. This year our Government ratified also the Convention on "Responsibilities in Case of Nuclear Damage";

- Exchange of experience on arrangements, relief, protection and recovery activities in the event of a major NPP accident in the territory of one country or transboundary contamination. Global experience is taken into account which makes the feasibility tests of the existing emergency plans much more reliable. A chance is offered also for perfection of the preparedness of the agencies and professional forces for population protection in nuclear accident.

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BG9700051

ANALYSIS OF HEAVY RADIOLOGICAL ACCIDENTS IN NPP AND AT GAMMA-IRRADIATORS

V. Angelov[1], T. Semova[2], Ts. Bonchev[2]

The harmless influence of the radioactive substances and their radiation over people is detected early in the 20th century but the society through the decades has accepted the described events as exotic ones and related to the activity of some dedicated scientists looking into one not so conceivable science. The problem became more topical when very frequent and strange diseases have been established among the staff servicing the roentgen equipment and this has led to the appearance of the first regulation of the activity that today we name as „work in ionizing radiation scope".

The two atomic bombs thrown over Hiroshima and Nagasaki in 1945 horrified the mankind but they were a result of deliberate activities at the time of a very hard world war. Announcements about victims of ground nuclear weapons testing appeared latter but this also did not concern the ordinary citizen in an ordinary state. Early in the 60th the global danger of a radioactive contamination arising from nuclear weapons experiments was recognized and beside the hard and complicated political situation the ground experiments have been ceased in practice.

Creation of nuclear-power engineering revealed for the first time a possibility of accident arising, i.e. an unplanned event which could lead unexpectedly to a new type of challenge of health and life of not suspecting anything people. All this at a time when the ways of an efficiently protection and the precise prognoses had not been known. Today the event of this type is called a „grave radiological accident".

Heavy accidents in industry are known since its development, especially in the chemical and military industries. The radiological accidents are something new and because of that and owing to the information media they are accepted by the people as an apocalypse threatening the human civilization. The simply comparison of the victim number of a radiological accident and all other accidents in one and the same period leads to another conclusion but here is not the place to discuss this problem.

Our task was to find out the literature sources concerning heavy radiological accidents, to show their properties and to make an effort to analyze the main causes of their arising. It turns out that the accidents of powerful gamma-irradiators represent not much lower danger to people than the NPP accidents, except for that in Chernobyl. For the sake of brevity the accident data are given in the form of uniform tables.

Table I. Heavy radiological accidents in NPP

No. 1

Object: Site: Accident date:	Heavy water experimental reactor NRX, thermal power 20 MW Chock-River, Canada 12 December 1952
Prime cause: Additional circumstances:	Wrong staff activities, pulling the core control rods At signaling „reactor emergency stop" not all rods fall gravitationally
Accident character:	Reactor heating up to 90 MW thermal power
Radioactivity emission	Reactor rooms contaminated of fission products with activity 10000 Ci
Irradiated people: Victims:	Staff Non
Economical damage:	Restored after 14 months
References:	[1]

[1]Civil Defence Administration, Sofia
[2]Sofia University „St. Kliment Ohridski"

No. 2

Object: Site: Accident date:	Plutonium production plant for military purposes Kishtim, North Ural 29 September 1957
Prime cause: Additional circumstances:	Failure in cooling system Nitrate-cellulose scrap explosion
Accident character:	Throwing great quantity of solutions consisting radioactive substances 11 μCi
Radioactivity emission:	Hard environmental contamination, approximately 15000 km^2
Irradiated people: Victims:	Non Approximately 10000 persons evacuated
Economical damage:	
References:	[2]

No. 3

Object: Site: Accident date:	Graphite-moderated reactor with air cooling for plutonium production Windscale, England 11 October 1957
Prime cause: Additional circumstances:	Wrong working conditions at graphite heating aiming internal stress take off Fire
Accident character:	Heat-eliminating elements melting
Radioactivity emission:	Hot particles and gases emitted in the atmosphere, including ^{131}I at total activity 20000 Ci
Irradiated people: Victims:	Staff and in some extent the population Non
Economical damage:	
References:	[1]

No. 4

Object: Site: Accident date:	Reactor at thermal power 500 MW NPP St. Laurent, France 17 October 1960
Prime cause: Additional circumstances:	At reactor in operation the refueling machine operator locks the automatic control system
Accident character:	Melting of part of the fuel
Radioactivity emission:	Non
Irradiated people: Victims:	Non Non
Economical damage:	
References:	[1]

No. 5

Object: Site: Accident date:	Heavy-water experimental reactor at low power Lusens, Switzerland 21 January 1969
Prime cause: Additional circumstances:	Technical trouble: corrosion of fuel shell
Accident character:	Explosion emitting approximately 1 t heavy-water
Radioactivity emission:	No emissions in the environment
Irradiated people: Victims:	 Non
Economical damage:	
References:	[1]

No. 6

Object: Site: Accident date:	Low-power experimental reactor (thermal 3 MW), SL-1 Idaho, USA 3 January 1961
Prime cause: Additional circumstances:	Involuntarily or purposely pulling out of control rods from the core in repair operation
Accident character:	Increasing the reactor power to 20000 MW in 0.01 s. Fuel melting which after reaction with the pressure vessel water leads to an instantaneous explosion resulting in roof throwing and its falling back again
Radioactivity emission:	High radioactivity in the working rooms
Irradiated people: Victims:	Three operators Three men dead
Economical damage:	
References:	[1]

No. 7

Object: Site: Accident date:	Brown Ferry NPP, three units, total power 1065 MW Alabama, USA 2 March 1975
Prime cause: Additional circumstances:	Plastic coating ignition by candle
Accident character:	Horizontally and vertically spreading fire leading to 2000 cables destruction
Radioactivity emission:	No emissions in the environment
Irradiated people: Victims:	No data Non
Economical damage:	10 bill. $ to repair. Two units not-working in 1 year
References:	[1]

No. 8

Object: Site: Accident date:	Three Mile Island NPP, one of the two units, power 961 MW every one Harrisburg, St. Pennsylvania, USA 28 March 1979
Prime cause: Additional circumstances:	Technical trouble: the operation of condensed system is ceased Water did not enter the steam generator as the valves to the auxiliary pomp have been closed since the last test
Accident character:	Core melting
Radioactivity emission:	Emission of noble gases and approx. 16 Ci Iodine to the atmosphere. High level of radiation in the reactor premises resulting of the noble gases
Irradiated people: Victims:	Staff and partially the population in the vicinity
Economical damage:	Required 1 milliard $ to repair but it is not restored
References:	[1]

No. 9

Object: Site: Accident date:	Gas-cooled reactor B2 Hunterstone NPP, Scotland 2 October 1977
Prime cause: Additional circumstances:	Corrosion leading to sea water rush in steam-generator
Accident character:	Increasing the humidity in gas heat carrier
Radioactivity emission:	Non
Irradiated people: Victims:	Non Non
Economical damage:	13 bill. ponds to repair, 28 months repair
References:	[1]

No. 10

Object: Site: Accident date:	Ginna NPP, power 490 MW New York St., USA 25 January 1982
Prime cause: Additional circumstances:	Tube destruction in steam-generator resulting in I circle water passing to II circle The prime cause is a forgotten disk with an weight of approx. 1 kg in I circle
Accident character:	The accident had been in control and no core melting occurred
Radioactivity emission:	Emission of negligible amounts of radioactive substances and noble gases to the atmosphere.
Irradiated people: Victims:	Staff - slightly Non
Economical damage:	
References:	[1]

No. 11

Object: Site: Accident date:	Chernobyl NPP, Unit IV, 1 GW Chernobyl, Ukraine 26 April 1986
Prime cause: Additional circumstances:	A sequence of operators failures in planned testing
Accident character:	Hydrogen explosion. Entirely distracted core. Fire with graphite combustion
Radioactivity emission:	Over 50 MCi - approx. 3.5% of the reactor radionuclide content at that time
Irradiated people: Victims:	Thousands of people and the population 32 persons
Economical damage:	Much milliards $
References:	[3,4,5]

Table II. Heavy radiological accidents in powerful gamma-irradiators

No. 1

Object: Site: Accident date:	Iridium irradiator containing 10 pellets at total activity of 35 Ci Houston March 1957
Prime cause: Additional circumstances:	Spilling of two pellets because of bad production in container repair operation The operators did not announce for the accident. It has been notified 1 month latter
Accident character:	Radioactive substance spilling
Radioactivity emission	8 houses and 7 cars contaminated
Irradiated people: Victims:	3 persons Non
Economical damage:	Restored after 14 months
References:	[6]

No. 2

Object: Site: Accident date:	Medical gamma-irradiator, ^{60}Co activity 450 Ci Ciudad Juarez, Mexico 6 December 1983
Prime cause: Additional circumstances:	Power gamma-source abandoned without control Source delivering to a metallic scrap melting workshop
Accident character:	Unsealing of a container consisting of 6000 pellets of ^{60}Co. Some pellets got into metal melting oven resulting in contamination of thousands of tons metallic production. Others have been spilled in yards, cars, etc. The contamination detection occurred 1,5 months latter
Radioactivity emission:	Contaminated melting ovens, workshop premises, dust-retaining systems, slugs
Irradiated people: Victims:	7 persons - 3-7 Gy; 73 persons - 0.25-3 Gy; 700 persons - 5-250 mGy Non
Economical damage:	
References:	[7]

No. 3

Object: Site: Accident date:	Medical Cs irradiator, activity of 1400 Ci at the time of accident Goiania, Brazil September 1987
Prime cause: Additional circumstances:	Irradiator abandoned without control after moving the clinic Gathering the source by metallic scrap gatherers
Accident character:	External irradiation and entirely unsealed ^{137}Cs source with specific activity of 15 Ci/g
Radioactivity emission	^{137}Cs contamination of large regions
Irradiated people: Victims:	2 persons - high doses, notified consequences latter in 500 persons 4 persons
Economical damage:	
References:	[8,9,10]

No. 4

Object: Site: Accident date:	^{60}Co gamma-irradiator, 18 kCi San Salvador 5 February 1989
Prime cause: Additional circumstances:	Locked removing sources system Spilling sources in an attempt to unlock the system
Accident character:	Operators irradiation which did not inform the administration. 4 persons more irradiated latter
Radioactivity emission:	
Irradiated people: Victims:	4 persons at high doses 1 person
Economical damage:	
References:	[11]

Table III Radiological accidents in Bulgaria

No.	Date	Object	Accident character	Causes	Irradiated
1	April 1980	LNT-Sofia	Premises contamination by ^{239}Pu. Maximum measured activity 3000 α-particles/cm^2	Careless staff work in a long time	
2	14.01.1983	IRT-2000 BAS	Premises contamination by ^{239}Pu	Operator's failure: solution of metallic ^{239}Pu instead of ^{235}U	Internal contamination
3	06.11.1985	SKTM Radomir	50 Ci ^{192}Ir source dropping at gamma-defectoscope refueling	Not made clear	Non
4	25.12.1985	„LATEX" Biala	^{239}Pu spilling of static electric neutralizers. 20 plates x 5 mCi	Fire	Not clear
5	27.08.1987	CLNT-CMI Sofia	Hand catching of a radioactive source, activity 0.07 Ci	Charging of ^{192}Ir defectoscope considering that inside have not radioactive pellets	8 Ber on the operator's fingers; total irradiation - 10 Ber
6	09.02.1988	HIMMASH Haskovo	^{192}Ir source dropping at gamma-defectoscope refueling	Mechanical failure in gamma-defectoscope	230 mR
7	05.10.1990	Opera in St. Zagora	Spilling of 50 fire annunciators with ^{241}Am and destructing of part of them, ^{239}Pu 0.5 mCi	Fire	Non
8	28.04.1992	Buhovo	Spilling of 18 ^{60}Co sources x 0.5 Ci on open ground	1. Delivered sources at activity of 1.5 Ci (1988) instead of standard sources 2. Careless removal of source container	
9	09.07.1992	Gas station Ihtiman	Source hold by hand, activity approx. 13 Ci. External irradiation of a people group	Falling out of a gamma-defectoscope source which has not been noticed	2 persons - 1-2 rad every one; 10 persons - 400 mrad every one

Some more serious accidents with powerful radioactive sources in Bulgaria

Some conclusions made in reviewing the accident data in Tables I and II are supported by the analysis of accidents with comparably powerful sources in Bulgaria. The main characterizations of these accidents are given in Table III using[12]. Radiological accidents arising in NPP „Kozloduy" are not included.

Main causes of heavy radiological accidents

The causes and characterization of the hravy radiological accidents in Tables I, II and III allow to draw the following conclusions:

1. Heavy radioactive accidents in NPPs have been connected with the human factor role and are due to:
   - flightiness, No.7;
   - carelessness, No8, No.10, No.4(?);
   - training, No.1, No.4, No.11;
   - deliberately, No.6(?);
   - rough technological failures, No.3, No.5, No.9.
2. The probability of arising and the heaviness of an accident depend on the technological failures which could be eliminated at better working devices diagnostics: No.2, N0.5, No.8, No.10.
3. The probability of heavy accident arising increases at time of reactor refueling, repair operations, planned testing: No.1, No.4, No.6, No.8, No.10, No.11.
4. In no one of the cases the plant did not have the readiness to meet and restrict the heavy accident development.

If now look at the accident with powerful gamma-irradiators (Table II) the conclusion could be filled out with:

5. Careless powerful radioactive source keeping: No.2, No.3.
6. Untrained staff to operate with such sources: No.1, No.4 and in Table III: No.2, No.5, No.8, No.9.
7. Fear of operators to announce the event leading to serious consequences: No.1, No.4.
8. The prime cause leading to the Buhovo accident (Table III No.8) has to be specially pointed out. At the time of ordering 18 sources of ^{60}Co early in 80th a change of activity unity occurred, from Curie (Ci) to Becquerel (Bq). This led to a mistake in the order and instead of standard sources of the order of mCi some sources of activity of 1.5 Ci have been delivered. Recognizing this the staff have abandoned the sources in a container from which in 1992 they have been spilled at wrong handling.

Some conclusions

The pointed out causes of heavy radiological accident arising itself prompt the conclusions. Nevertheless we will discuss some of them in details.

It has no doubt that the main cause of a heavy radiological accident arising is the human factor. At first sight the role of this factor would be as lower as is the staff qualification higher. Indeed the gravest accident, the Chernobyl accident, could not arise if the staff recognized the processes in the reactor. The same is the case at Three Mile Island NPP accident. In cases where there is a carelessness and even flightiness this could not be referred only to the qualification. Unfortunately in the literature we have in our possession there is not an analysis of the behavior of people causing an accident. Even an high-skilled specialist is capable of such activities in certain circumstances: overwork, personal troubles, lack of sufficient training etc. It is well-known that the reactor operators spend much of the time observing a variety of instruments being in continuous psychological strain. This lead to some neurosis and psychosis to people with high intellectual properties.

Currently „Kozloduy" NPP is an object of interest and supervision of various external organizations and first of all of IAEA. No doubt that these institutions contributed and contribute to the elevation of plant working level and security. It is an important fact that in official an public attitudes these missions give a high evaluation of the professional properties of our specialists in „Kozloduy" NPP.

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ON SOME PROBLEMS CONCERNING THE NATIONAL EMERGENCY PLANNING

V. Angelov[1], Ts. Bonchev[2], Sv. Andonov[1], T. Semova[1] .. Gantchev[3], V. Georgiev[4]

In the middle of 60th the nuclear energetics began to play an important role in meeting the need of electric power of the society. During the past period in a number of nuclear energy enterprises some events occurred of which three are substantial: the Windscale processing mill accident (1957), the Three Mile Island NPP accident (1979) and the Chernobyl NPP accident (1986).

Every one of these accidents led to considerable changes in the development of the radiological accident response concept for protection of staff and population and of the emergency planning criteria in a national and international aspect. The large-scale accidents in definition are accidents in which the control over the irradiation source (respectively the nuclear reactor or its crucial systems) is lost and the commonly accepted limits of dose restriction at normal life recommended by international and national bodies become inapplicable. The protection of staff, population, national economy and environment in the „emergency planning area" is implemented in any form of intervention. Such an intervention is regulated in advance and is the foundation of the NATIONAL EMERGENCY PLANNING AND PREPAREDNESS.

After the Windscale (now Selafield, England) accident[1] it had been accepted in an international aspect that the different forms of intervention have to be selected on the basis of the prognostic dose burden. The Windscale experience showed that the well-planned, practical and regularly repeated emergency preparations had been recognized in an early stage. So during the 70th a term of the license for operating nuclear reactors was to have provided emergency preparations. Since than these preparations have been continuously reviewed and in the course of time many detailed improvements have been introduced as a result of the training experience and in close relation with other bodies participating professionally in the process. The Three Mile Island NPP (USA) accident[2] led to some changes in the concept grounding two levels of intervention: one at which the countermeasures are introduced by estimation, and the other at which they are binding. The numerous lessons learned at Three Mile Island resulted to the introduction of new measures ensuring the safety of nuclear energy equipment even though related to expense increase.

The Chernobyl accident[3] had an influence over the emergency planning principles both on national and international levels. Special attention is paid to the emergency planning and preparedness not only in national but in regional scale on the basis of signed bilateral and multilateral conventions of the way of annunciation, coordinated activities of accident localization and settlement, international aid etc.

Basic principles of emergency planning and preparedness

The principles of emergency planning and preparedness as used in some developed countries as USA, England, Sweden, Belgium, France etc. are as following:

1. Availability of a legislation of atomic energy use and especially a legislation in the field of population protection in case of an accident including an radiological accident.

2. Availability of continuously operating body for emergency planning and up-dating of the emergency plans.

3. Availability of a national system of radioactive contamination control for the country.

[1]Civil Defense Administration, Sofia
[2]Sofia University „St. Kliment Ohridski"
[3]Commitee for Peaceful Use of Atomic Energy
[4]ENERGOPROECT

4. Communication system translating the whole information in case of an accident, most frequently this is a military communication system which in peacetime has redundant free capacity.
5. Scientifically grounded determination of emergency planning areas.
6. Scientifically grounded determination and application of countermeasures for population protection.
7. Availability of a efficient system for population announcement.
8. Availability of evacuation plans in case of probable contamination and real radioactive contamination.
9. Scientifically grounded levels of intervention (criteria for decision making of population protection measures application).
10. Organizational preparedness of the local authorities.
11. Lawfully regulated responsibility for population protection including in case of a nuclear accident on the country territory and at transboundary transportation of radioactive substances.

The national emergency planning and preparedness consist of inter-related and crossed plans of various organizations each of which has its specific responsibility scope. An important property of accident prognosis in the National emergency plan[7] is that all projected models are not strictly restricted to certain kind of event rather outlines the frame of a flexible response whose detailed characterizations could not be foreseen. It is also very important those who participate in the emergency response to recognize where their responsibilities begin and where they end. The Permanent Commission of Population Protection at Disasters and Accidents to the Council of Ministers of Republic of Bulgaria[8] is charged with the ultimate responsibility to make adequate decisions relying to its expert commissions and many other professional organizations. A wide spread of the population protection activities organization at an eventual accident is published in a detailed emergency manual[9] which has certain importance in the post-Chernobyl era.

As a whole the National emergency plan is developed in accordance with the present legislation and the available national structures and organizations for operations are maximum utilized. This is in accordance with the signed at present international conventions and agreements and the current requirements of IAEA[10].

The National emergency plan has to be quickly specified and coordinated with corresponding ministries and departments which would accelerate its identification. This would improve its readiness for activity and would create conditions for future development.

The basic recommendations could be summarized as follow:

- Currently legislation does not provide corresponding frame base therefore a new legislation is required. The new legislation would outline the necessities and the scope of the national emergency planning in the part concerning NPP „Kozloduy".
- Civil Defense must be separate from the Ministry of Defense rending an account directly of the Ministry Council. Civil Defense has to become a national institution responding to accidents, responsible for developing and coordination of the national emergency planing and preparedness.
- The Scientific-coordination council[11] to the Permanent Commission of Population Protection at Disasters and Accidents to the Council of Ministers of Republic of Bulgaria has to work out a „Technique for prognosis and assessment of the

radiological situation at accidents in NPP „Kozloduy" or at transboundary transportation of radioactive substances".

- Drill program of the National emergency plan has to be worked out so that to vouch for a sound plan testing and efficiency of exercises of emergency staff. This program has to range over exercises, training, headquarters drills in restricted scale and in large scale. The supply of these drills has to be provided by the new legislation.
- It is required an annual reviewing of the finance for technical service up-dating and repairing in the emergency area.
- To avoid some inadequacies, especially after every up-dating, and in order to coordinate the emergency planning efforts on and out of the sanitary-protection zone of NPP the procedures of the National emergency plan and these of the NPP „Kozloduy" emergency plan have to be synchronized.
- The Permanent Commission of Population Protection at Disasters and Accidents to the Council of Ministers of Republic of Bulgaria has to accelerate in priority the creation of a functional structure of a „National system of radiological control protecting the population, environment and the national economy" providing resources and using the staff potential and equipment in the country for radiological protection.

An important characterization of every emergency planning is that it has to be weighed according to the real conditions in the country. This is supported by the principle of an American expert calling it „national level of ambition"[13]. This principle reflects the country potential capabilities of introducing intervention levels and permissible dose burden. In our country this is not done and has to be done in future in the emergency protecting planing.

The acceptance of the above mentioned recommendations will provide a suitable medium of efficiency emergency response as well as the infrastructure of its support. Having in mind that not all of the pointed out recommendations are reliable at the moment it is important some procedures for their gradually achievement without technological, administrative and finance obstruction to be developed.

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INFLUENCE OF NPP „KOZLODUY" ON THE HEALTH STATE OF PERSONNEL AT A NORMAL OPERATIONAL REGIME

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This paper sums up the results from the medical observation on the NPP personnel for 20 years period of time - 1974-1993. Data from the medical service at NPP, from "Radiation Medicine" laboratory at NC' 'P and from other laboratories in the Centre is used.

NPP personnel is exposed to the combined influence of radiation and non-radiation factors of the working sphere. From the radiation factors of importance is external exposure, realized mainly by the gamma-radiation of radioactive substances - products of division and activation. Internal exposure is formed chiefly out of cobalt 60, caesium 137 and caesium 134, dose from external exposure being under 10% of the annual dose limit (ADL).

The number of the persons controlled in NPP is 3315. 1804 of them work in EP-1 and EP-2, and 1511 persons work external organisations. The number of irradiated persons for 1993 is 2881 (87%). Dose exposure for 1993 is distributed quite irregularly: 43% of the workers have received dose under 1 mSv and only 2% - from 30 to 50 mSv. The per cent distribution of workers depending on the doses received during 1987-1992 period is presented in Table 1.

The cumulative exposure dose of personnel varies from 50 to 620 mSv for their whole working period in NPP. The highest doses are received by the workers dealing with repairs, the deactivation groups follow them. NPP management receives the lowest doses.

The prophylactic medical examinations include examinations carried out by different specialists, as well as different types of studies; they are shown in Table 2.

The health state of NPP personnel for many indicators is compared with TPP or with summed up data for the country. It is viewed in the following aspects:

- Disease incidence with temporary incapacity for work
- Analysis of results from the prophylactic examinations
- Results from the medical examinations on risk groups of workers

The frequency of cases with temporary incapacity for work varies from 200 to 600 % at 800 - 900% for the country. The main reasons for the low disease incidence with temporary incapacity for work are due to the fact that the personnel is chosen by strict medical criteria and therefore has a better initial health status. Moreover, working in ionising radiation conditions, NPP personnel uses a number of compensations and social acquisitions, which have a favourable effect on the health status.

The frequency of lost days due to temporary incapacity for work is about 400 % and is lower than that for the group of metallurgists, miners and the capable-of-working population in the country as a whole.

The average duration of incapacity for work of one case in NPP is 12.9 days, and in TPP "Sofia - Iztok" - 7.7 days. The longer duration of the incapacity for work in NPP is due to the presence of cases needing a longer treatment, infection diseases, for example, which come 2nd or 3rd in frequency of cases.

First come acute infections of the upper respiratory tract, home accidents, diseases of the digestive system and of the autonomic nervous system, etc.

The prophylactic medical examinations of NPP personnel are carried out in accordance with the normative documents of the Ministry of Health.

About 98-100% of the workers are examined prophylactically every year. Figure 1 shows the distribution of diseases on 1000 persons. It should be noted that in this figure, "hypertonia arterialis" (group 2) includes also the cases of dystonia neuro -circulatoria of a hypertensive type, and that in eyes diseases (group 10) are included also the optical anomalies, which in most of the cases are innate (daltonism, for example, changes in visus, etc.)

The distribution according to ages shows a maximum of the diseases in the group of 41-50 years-old. There is a certain age dependency in some groups of diseases. This dependency is similar to the total picture in the country. For example, the gastric and duodenal ulcer is observed more frequently in the 31-40 years-old, neuroses - maximum for 41-50 years-old, diseases of the respiratory system - 51-60 years-old, etc.

The distribution of diseases according to length of service shows increased disease prevalence for the workers with a longer service in NPP. This, however, correlates in most of the cases also with the increase in age. No dependency between exposure cumulative dose and disease prevalence level is observed.

During the 20 years period of examination on NPP "Kozloduy" workers no cases of specific radiation pathologies have been registered. This corresponds to the levels of personnel occupational exposure and to the lack of accident situations due to overexposure.

Most detailed examinations are carried out on risk groups of workers. These are workers who possess a relatively higher cumulative exposure dose or who have received annual dose over the annual dose limit of 50 mSv (they are 9 persons with length of service of 13-16 years, workers aged up to 35 with cumulative doses over 300 mSv, women at reproductive age working in the control zone.

One of the most sensitive biological indicators of ionizing radiation influence are peripheral blood indicators. They are within the limits of the referential values. The peripheral blood leucocyte count in dependency of the cumulative dose is presented in Fig.2. The biochemical studies are normal too. The immunological studies (cellular and humoral immunity) show variety, being within the limits of the referential values. The dynamics of immunoglobulins in dependency of cumulative dose is presented in Fig.3. Leucocytes phosphatase alkaline and myeloperoxidase tend to increase, however they do not exceed the limits of control values (Fig.4 and Fig.5). The observed changes are taking the course of the general adaptation syndrome and they are not contra-indications for work in NPP.

Conclusions:

1. Disease incidence with temporary incapacity for work of NPP "Kozloduy" workers is lower than the one for TPP and for the summed-up data for the country.

2. The structure of general disease prevalence is determined mainly by diseases of the respiratory, digestive, nervous and locomotor systems. This structure is typical for most of the industrial productions in the country.

3. The dynamic medical observation on NPP "Kozloduy" workers has not shown any cases of radiation injuries. This is due to the lack of accident overexposures of personnel and corresponds to the levels of the received cumulative exposure doses.

4. Peripheral blood indicators, as well as the results from biochemical and cytochemical studies are within the limits of the referential values.

5. A variety in the indicators of immune and endocrine systems as an adaptation reaction to working conditions factors is observed.

6. The changes registered in katheholamines concentrations for a part of the operating-repairs personnel are manifestation of tension in adrenaline-sympathetic system of constitution during work. Additional studies are necessary in this case.

7. The health state of NPP "Kozloduy" personnel is very good as a whole and does not indicate any specific pathology. It corresponds to the working conditions in the NPP and to the given data for personnel health state in other NPPs in the world.

8. Special attention has to be paid to the risk of development of late effects for personnel health regarding the preliminary results for increased frequency of chromosome aberrations. More detailed epidemiological studies are necessary in this case.

Table 1. Percent distribution of persons in dependency of doses received for 1987-1992 period

under 1.0 mSv	30.9%
1.1 ÷ 4.0 mSv	27.9%
4.1 ÷ 12.0 mSv	21.8%
12.1 ÷ 30.0 mSv	13.4%
30.1 ÷ 50.0 mSv	5.8%
over 50.1 mSv	0.2%

Table 2. PROPHYLACTIC MEDICAL EXAMINATIONS ON NPP WORKERS

clinical examinations by:	tests:
internist	FBP with DBP
radiobiologist	biochemical
neurologist	cytochemical
ophthalmologist	immunological
otorhinolaryngologist	personal radiation sensibility
operating surgeon	radioimmunological
gynaecologist	radiochemical
psychologist	electrophysiological
dentist	cytogenetic
	urine tests
	psychophysiological, etc.

Terms for carrying out:

for CZ - once in 6 moths

for the rest of personnel - once a year

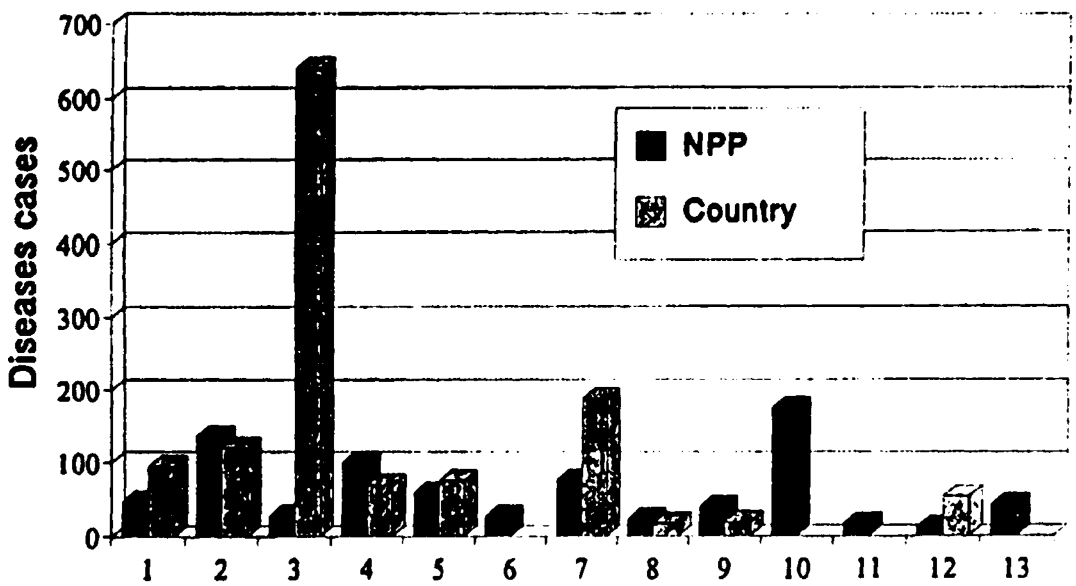

Fig. 1: Diseases of staff in Nuclear Power plant „Kozloduy", compared with population in Bulgaria (on 1 000 person)

1 - Cardiovascular system
2 - Hypertonia arterialis
3 - Respiratory system
4 - Digestive system
5 - Gynaecologic diseases of women
6 - Kidney - sex system
7 - Neurologic system
8 - Endocrinologic system
9 - Obesitas
10 - Eyes
11 - Haemopoietic system
12 - Bone - muskulative system
13 - Other

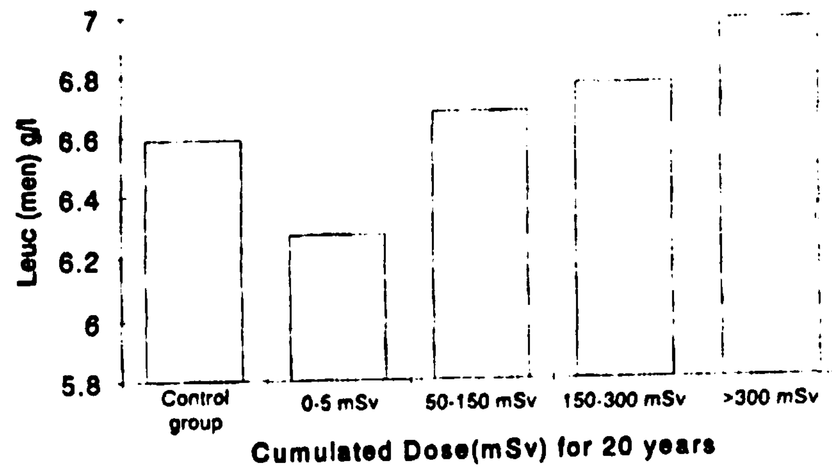

Fig. 2: Change in periferial blood leucocyte count in dependancy of the cumulative dose

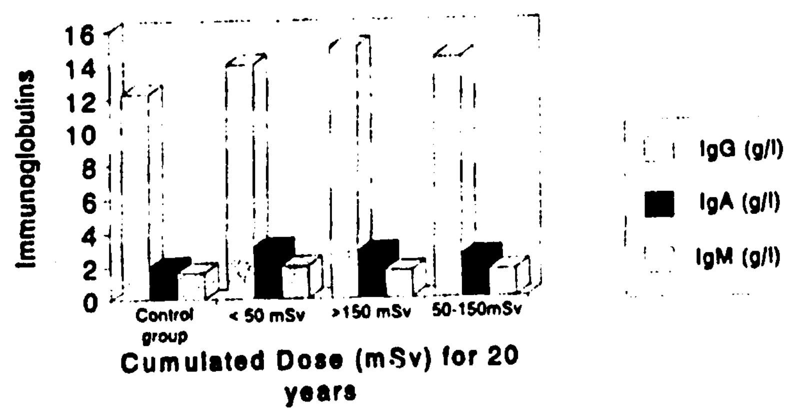

Fig. 3: Immunoglobulins in periferial blood in dependency of cumulative dose

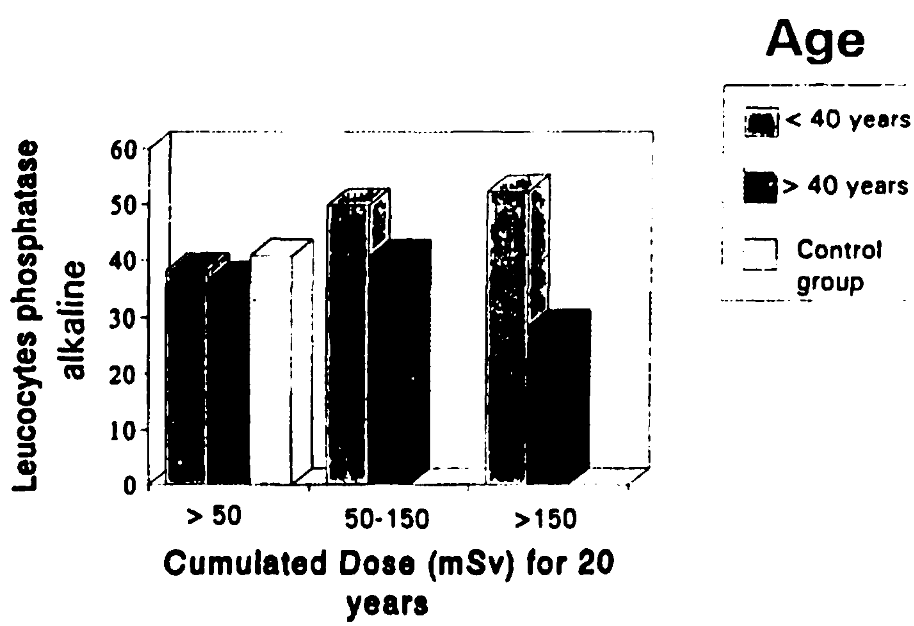

Fig. 4: Leucocytes phosphatase alkaline in periferial blood in dependency of cumulative dose

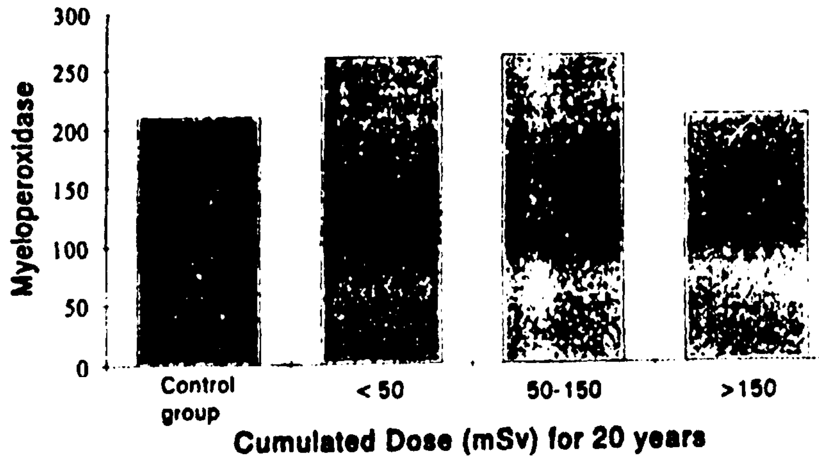

Fig. 5: Myeloperoxidase in periferial blood in dependency of cumulative dose

BG9700054

A CYTOGENETIC STUDY OF PERSONS WORKING AT THE NUCLEAR POWER PLANT „KOZLODUY" WITH A VIEW TO THE HAZARDS OF LATE EFFECTS

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The normal „accidentless" functioning of nuclear power plants (including the NPP „Kozloduy") is accompanied by radiation exposures of workers in the range of the low doses. The risk of the occurrence of health consequences due to the radiation factor concerns mainly late radiation effects which are stochastic (without a threshold) malignancies manifested after various periods of time or genetic defects in progeny. Their possible occurrence is most genuinely prognosticated on the basis of cytogenetic investigations on workers. An established fact in modern science is the detection of chromosome damage as an event preceeding the mentioned effects. Besides, the status of the chromosomes in persons frequently engaged in activities with an increased risk of radiation exposure illustrates the efficatiousness of the security precautions in the best possible way. It should be noticed that the chromosome lesions analyzed by the conventional method, are not used as a biological dosimeter after a long-term (plolonged or chronic) radiation exposure opposite to cases of acute (incidental) exposures. They are considered an indicator of a realized damage which depends not only on the exposition dose but also on a series of individual peculiarities the genetic nature, harmful habits, life style, etc.

The present study includes investigations on the chromosomal status of 40 workers from the NPP "Kozloduy" by using 3 cytogenetic end-points with a different information value which are considered most suitable for biomonitoring in genotoxicology, namely: chromosome aberrations (CA), sister-shromatid exchanges (SCE), micronuclei (MN) in peripheral blood lymphocytes. This study is a part of a planned program of medicogenetic investigations on risk contingents consisting of the workers of the zone of strict control at the NPP „Kozloduy" with a view to a risk assessment of the late radiation effects and the possibilities of carrying out an appropriate prophilaxy. At the present stage the investigations range over workers whose occupational conditions predispose to a relatively higher radiation exposure. 40 workers were studied, 32 of them working at the Ist IVth unit, 5 of them at the Vth VIth one, 2 at CPRR and one is the head of the repair department. Their average age is 39,9. About half of them belong to the group of persons with 11-15 years of service, 25% have served for 6-10 years, and 24% for 16-20 years. Concerning their exposure, 30% of the workers have accumulated a dose of 20-30 cSv, 27,5% between 30-40 cSv, 25% between 40 and 50 cSv, one person has accumulated a dose in the range of 50-60 cSv, one person has been with a dose lower than 5 cSv. 2 of the persons have not been subjected to a dosimetric control. During 1993 3 of the workers have been exposed to a dose higher than 3 cSv and 11 to a dose over 2 cSv for a period of 9 or 12 months. The doses of the rest have been lower than 2 cSv for the mentioned period in 1993

Each person has been inquired in detail into his or her occupational, life-style, health or family anamnesis. The investigations were completed at the earliest several months after taking the samples and the results have been sent to the health service of the plant. In many cases an appropriate information has been given to the investigated persons.

Investigations by the chromosome-aberration (CA) assay in peripheral blood lymphocytes.

This assay is considered the most sensistive and the most illustrative one concerning a realized chromosome damage after a mutagenic impact. Persons with an increased frequency of chromosome aberrations (especially when repeatedly proven) are regarded as a risk contingent chracterized by a greater probability for the occurrence of malignancies or genetic defects in progeny in comparison with the rest population. This assay is considered most informative in case of investigations on persons occupationally exposed to radiation. The fulfilment of the assaying has been carried out in compliance with the international protocols. The results are presented in Table 1. The frquency of cells with chromosome aberrations (as well as the total number of aberrations) is significantly increased in the group of workers from the NPP „Kozloduy" compared to the controls. The frequency of the dicentrics is also increased and this kind of a chromosome aberration is considered, to a great extent, specifically induced by the radiation factor. In the group of workers from the NPP, the distribution of persons with an increased frequency of chromosome aberrations has been calculated by the method of signal deviations as follows: 45% of the persons show the pointed mean value 2,46%, the frequency is higher in 25% and in 30% it is lower than the mean one, i.e. a higher yield of chromosome aberrations is found in 70% of the investigated workers in comparison to the spontaneous frequency. Dicentrics are detected in 27,5% of the investigated persons. The age of the staff is of significance when compared to that of the control group since the spontaneous frequency of chromosome aberrations is usually increased as the age grows. In this case persons at the age of 20 up to 60 years are included in the control group and their average age is higher than that of the workers 39,9. A direct relation is not found to the accumulated dose but the group is too small to make an explicit conclusion. 2 of the 12 persons with an unincreased frequency of CA have accumulated a dose varying from 40 to 50 cSv, 4 from 30 to 40 cSv, 5 from 20 to 30 cSv and 1 has been exposed to 1,95 cSv. The influence of additional unfavourable factors has been analyzed too, such as preceeding diagnostic procedures with the use of ionizing radiations, a contact with chemical noxes and smoking. A direct relation was not found between these agents and the frequency of CA.

Investigations by the micronuclear (MN) assay in peripheral blood lymphocytes

This is a screening assay which is considered suitable to prove the mutagenicity of a certain agent. It does not show any specificity with respect to ionizing radiations. It defers to the metaphase chromosome assay on its information value and sensitivity but the MN assay is significantly less labour-consuming.

The assay is carried out according to he international protocols. 2000 cells are analyzed from each person for 28 workers, 1500 for 2, and 1000 for 10 workers. The results are presented in Table 2. The detected number of micronuclei is 20,9 on the average in peripheral blood lymphocytes from workers of the NPP „Kozloduy". Since the control group of our laboratory is not representative enough the juxtaposition has been made on the basis of the available published data. The frequency of MN in the group of workers from th NPP „Kozloduy" is increased when compared either to the control one or to the sponataneous frequency reported about numerous groups belonging to the Australian and the Hungarian populations (Table 2) (4,5). However, the data presented by Vaglenov et al. on control groups of the Bulgarian population show rather high MN values and according to them no increase is observed in our investigations (1). The average group value found by us is valid for 50% of the workers, 17,5% of them are characterized by a higher one, and 32,5% by a lower than the average value. It can be finally concluded that, at this stage od the study it is not possible to affirm the increase of the MN assay in the investigated workers' group. It is necessary that we should extend the investigations by using both assays.

Investigations by the sister-chromatid exchange (SCE) assay in peripheral blood lymphocytes

The SCE assay is used routinely in the cytogenetic monitoring of persons exposed to an intensified mutagenic impact. It is considered that the SCEs are increased mostly under the influence of chemical mutagens but it is quite possible to find the assay weakly posistive after low-dose exposures to ionizing radiation. The assay is regarded as one of the most sensitive to the impacts of tobacco-smoking on the genetic structures.

The assay has been realized according to the international protocols: 30-50 cells have been analyzed from each of the investigated persons. The obtained results are shown in Table 3. The determined average frequency of SCEs is about 2,5 times higher than that of the control group. Most probably, this might be due to the great number of intense smokers among the workers. 67,6% of them are current smokers while 92% were used to smoking in the near past. The observed discrepancies, i.e. such as an increased frequency of SCEs in case of 3 non-smokers as well as an unincreased frequency for six workers ,from whom only 2 are non-smokers at present, are single cases which do not allow any conclusions. No correlation is observed between the frequency of CA and SCEs. The search for a relation between the SCE frequency and incidental contacts with other chemical noxes does not give evidence of any dependency. It could be accepted that tobacco-smoking potentiates the activity of the basic mutagen (in this case the low-dose radiation exposure).

The detected increase of CA among the investigated workers from the NPP „Kozloduy" is of values similar to those found by Belgian, English, and German authors for other NPPs (5,7,11,12). Comparing our results to the reported frequency of CA for workers from thermoelectric power plants, nuclear reactors, industries with chemical noxes (oil products, pesticides, chemicals, solvents, rubber, etc.), it could be concluded that the risk of the incidence of late effects for the investigated persons from the NPP „Kozloduy" could be included into the generally accepted risk for persons working under noxious conditions (2,4,9,13,14,15).

CONCLUSIONS:

1. A higher frequency of chromosome lesions has been detected for the investigated group of 40 nuclear power plant workers from Kozlodui compared to a control group of the Bulgarian population of which no data is available of an addditional mutagenic impact.95% of the workers have been employed for more than 5 years at the NPP"Kozloduy" and 60% have accumulated a dose of more than 30 cSv. An alarming fact is the comparatively young average age of the investigated persons, i.e. 39.9.

2. The detected chromosome damage is most probably due to the radiation factor taking into consideration the frquency of a kind of the chromosome aberrations dicentric chromosomes which are reckoned to be, to a great extent, specifically induced by radiation exposures.

3. There is no reliable evidence of a direct personal dependence of the chromosome-aberrations frequency on the accumulated dose.

4. The wide currency of tobacco-smoking among the investigated nuclear-plant employees potentiates additionally the damage of the chromosome structures caused by ionizing radiation exposures.

RECOMMENDATIONS:

1. The detected damage reqires a regular specialized control over the investigated persons. According to the world standards persons with a repeatedly proven higher frequency of chromosome aberrations are regarded as a contingent with an increased risk of the occurrence of ning their preventive removal from the
working places.

2. All persons signing on to work in the controled zone should be informed about the probability of increased hazards of late-effect occurrence. Such a risk is inevitable for all occupations with additional mutagenic impacts and it cannot be discounted in nuclear-power plants everywhere.

3. The conditions of work in the Ist unit should be analyzed and all possible precautions undertaken to enhance the protection against radiation.

4. It is necessary to carry out a prophylaxis spreading over the persons with an increased frequency of CA and correspoding to modern knowledge of the relationship between mutagenic and carcinogenic processes. The preparation of prophylactic measures should be in conformity with the world practice.

5. A cytogenetic mopnitoring should be carried out among all persons belonging to risk contingents. The data on persons currently working in the Ist unit should be compared to those of the employed in the Vth-VIth unit where the level of radiation exposure is low.

6. Strict precautions are to be taken for the observation of the no smoking ordinance in the controled zone and a comprehensible popular persuasion is needed which elucidates the specific harmful effect of tobacco smoking on nuclear-power plant workers.

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TABLE 1.

CHROMOSOME-ABERRATIONS FRQUENCY IN WORKERS FROM NPP "KOZLODUY"

Groups	Number of inv. persons	Cells with chromosome aberrations % ±SE	Total of chromosome aberrations* % ±SE	Dicen trics % ± SE
Workers from NPP"Kozloduy"	40	2,46±0,2	2,65±0,2	0,25±0,1
Length of service/years/				
2-5	2	3,0	3,25	0,25
6-10	9	2,3	2,5	0,11
11-15	18	2,3	2,9	0,33
16-20	8	2,2	2,2	0,11
Accumulated dose /cSv/				
No dosimetry	2	2,5	3,5	0,25
up to 20	2	2,7	3,2	1,50
20-30	12	3,0	3,3	0,04
30-40	11	1,6	2,4	9,14
40-50	10	2,2	2,3	0,35
Over 50	3	3,8	3,8	0,33
CONTROLS	173	1,56±0,03	1,56±0,03	0,06±006

TABLE 2.

INVESTIGATIONS ON WORKERS FROM NPP "KOZLODUY" BY THE MICRONUCLEUS /MN/ ASSAY

MICRONUCLEUS ASSAY/MN/			
Groups	Number of investigated persons	Micrinuclei per %±E	Cells with micronuclei %±E
Workers from NPP "Kozloduy	40	20,9±2,2	18,1+1,8
Controls -Bulgarian population /by the same authors/	6	13,5±2,0	12,25±2,0
-Bulgarian population /by another author/	17	23,7±4,6	
	8	24,3±4,0	
	4	27,5±5,8	
-Australian population	225	17,2±2,0	
-Hungarian population	188	16,0±1,2	

TABLE 3.

INVESTIGATIONS ON WORKERS FROM NPP "KOZLODUY" BY THE SISTER CHROMATID EXCHANGE /SCE/ ASSAY

Groups	Number of investigated persons	SCE per cell /standard deviation/ (a)
Workers from NPP "Kozloduy"	40	18,9 /6,5/
Smokers	25	19,2 /6,9/
Non-smckers	12	18,0 /6,3/
Smokers in the past	30	18,8
Controls	38	7,3 /1,5/

BG9700055

ANALYSIS OF DOSE DELIVERY PATTERNS TO „KOZLODUY" NPP PERSONNEL

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The object of this report is to describe evolution of some basic characteristics of occupational exposures during 1974 - 1993 period.

The information for this report was obtained from authorities of NEC NPP „Kozloduy" as well as from NCRRP* monitoring carried out in 1993.

Reactor units

Now 6 reactors from type PWR** are in operation in NPP „Kozloduy" - four of 400 MWe type and two of 1000 MWe type. A chronology of setting them into operation is represented at Table 1.

Table1. A chronology of setting PWRs in NPP „Kozloduy"

Year	Small plants 400 MWe	Medium plants 1000 MWe
1974	1	-
1975	1	-
1981	1	-
1982	1	-
1988	-	1
1991	-	1
Total	4	2

Nuclear power plant experience reaches 73 reactor year for the 6 reactors. The average age per reactor is respectively 16 years for 400 MWe and 4.5 years for 1000 MWe PWRs.

In the last year the oldest two reactors were backfitting modificated under surveillance at VANO experts. The same is going to be performed whith other two 400 MWe PWRs.

Annual collective dose

The annual collective dose increases whith the number of reactors (Table 2). The total collective dose cummulated in NPP „Kozloduy" since the begining of Bulgarian Nuclear Power programme up to 1992 reaches about 165 manSv.

In 1993, the two 1000 MWe reactors represent 17% of the total collective dose for all six PWRs.

Collective dose per GWh

At the seventies collective dose per unit of electricity produced amounted to about 1.2 mmanSv per GWh (table 2). This were the first year of experience in use of nuclear power instalation. At the eighties, the ratio of mmanSv per GWh decreased and was held in limits of 0.6 - 1.0.

Figure 1 shows a comparison of this index for NPP „Kozloduy" and OECD*** regions.

Annual average collective dose per reactor

At the seventies and eighties, annual average collective dose per reactor changed approximately between 2 and 3.5 manSv.

After 1987 the annual average collective dose per reactor decreased and was held at a value lower than 2.0 manSv.

This could be explained by the exploatation of more modern 1000 MWe PWRs.

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* NCRRP - National Centre of Radiobiology and Radiation Protection
** PWR - Pressurized Water Reactor
*** - Organization for Economic Co-operation and Development

A comparison of this index for NPP „Kozloduy" and OECD countries is shown at figure 2.

Personal annual doses

The summarized data of personal annual doses registrated for the period 1987 - 1992 are represented at Table 3. The data concern the whole staff employed in maintenance and exploatation at NPP „Kozloduy". The participation of other organizations is about 20%.

According to Table 3 68.6% of the registrated personal annual doses are up to 4 mSv/a, 85.4% - up to 12 mSv/a, 99.8% - up to 50 mSv/a.

Comparatively higher is the dose of the personnel occupied in reactor-repair operations and outage of the reactors.

The average personal annual dose during this years is in the range of 4 - 8 mSv/a.

For 1993 the average personal annual doses are as follow: 1.3 mSv/a for the personnel of 1000 MWe PWRs and 5.5 mSv/a - for the personnel of 400 MWe PWRs.

Conclusions

1. The total collective dose cummulated in NPP „Kozloduy" since the beginning of the nuclear programme up to 1993 reaches about 165 manSv (73 reactor - years) for all the six PWRs.

2. The index of the annual average collective dose, per reactor in NPP „Kozloduy" correlates whith OECD countries and even is better than that at Germany, USA and Spain.

3. The index of the average collective dose per GWh in NPP „Kozloduy" correlates whit that of USA, but is at unfavorable levels than European countries indexes.

4. The undergoing novation of the 400 MWe PWRs is a circumstance for improving the above indexes.

5. During 1987 - 1992 period the represented personal annual doses are as following: 68.6 % - up to 4 mSv/a, 85.4 % up to 12 mSv/a, 99.8 % - up to 50 mSv/a.

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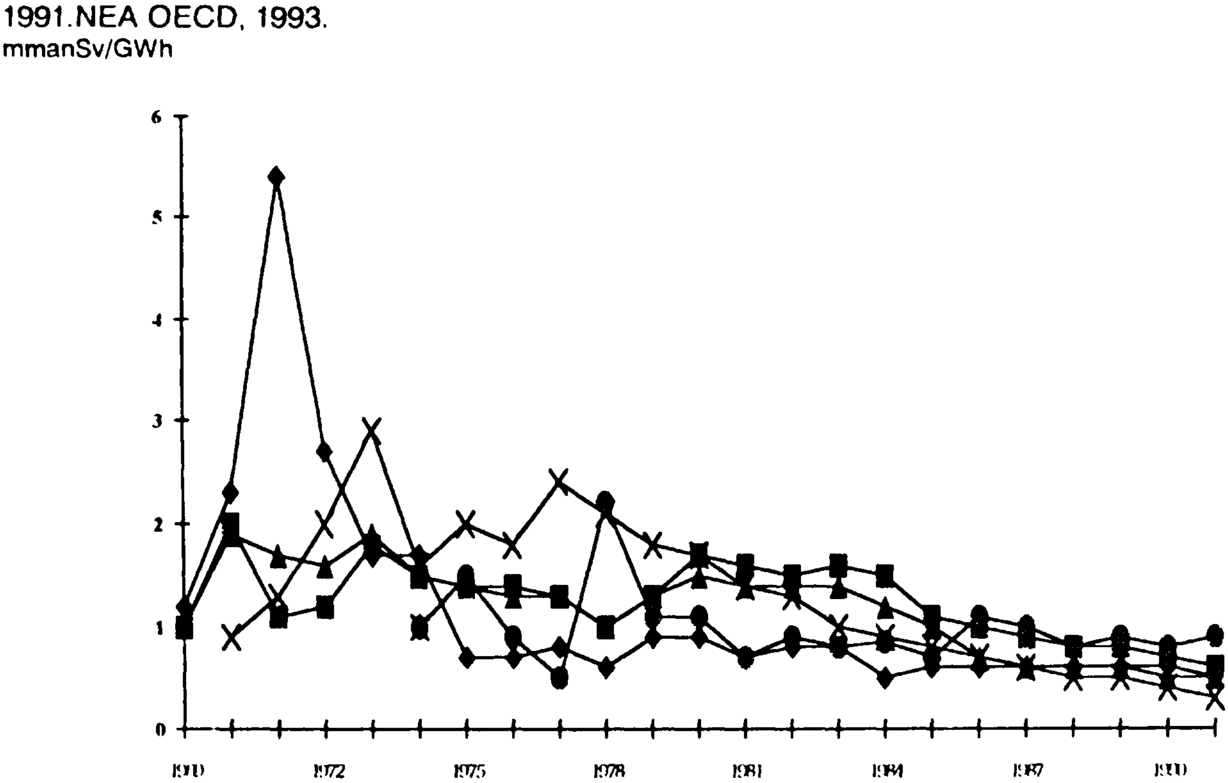
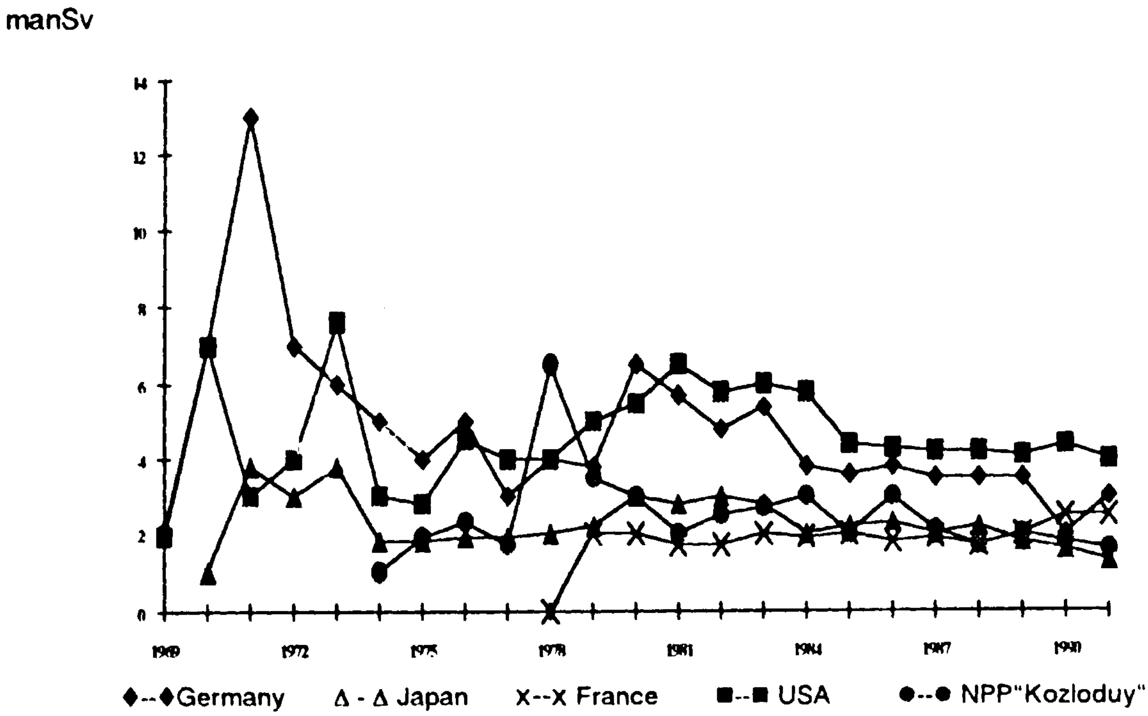


◆--◆Europe ■--■ America x--x Asia Δ-Δ All OECD Reactors ●--● NPP Kozloduy

Figure 1. Average collective dose per DWh for NPP "Kozloduy", Bulgaria end OECD regions

a)



b)

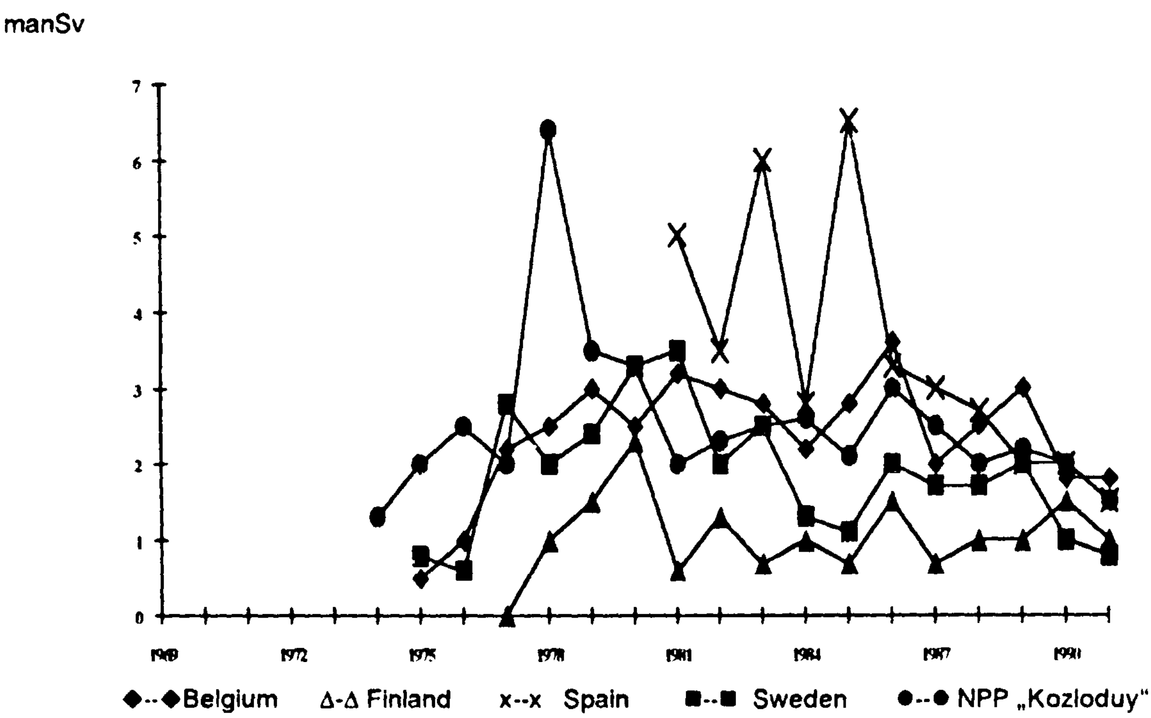


Figure 2: Annual average collective dose per reactor for NPP „Kozloduy"
Bulgaria and OECD countries:
a) Germany, France, Japan, USA
b) Belgium, Spain, Finland, Sweden

Table 2: Some data of NPP "Kozloduy", Bulgaria for 20 years opperational period

year	Number of reactors	Gross production (GWh)	annual total collective dose (manSv)	Average collective dose per GWh (mmanSv/GWh)	Average collective dose per reactor (manSv)
1974	1	928	1.07	1.15	1.07
1975	2	2555	3.87	1.50	1.91
1976	2	4989	4.72	0.95	2.36
1977	2	5884	3.62	0.62	1.81
1978	2	5911	12.60	2.13	6.30
1979	2	6165	7.03	1.14	3.51
1980	2	6165	6.79	1.10	3.40
1981	3	9119	6.29	0.69	2.10
1982	4	10846	10.01	0.93	2.50
1983	4	12317	10.28	0.83	2.57
1984	4	12735	10.58	0.83	2.65
1985	4	13131	8.21	0.63	2.05
1986	4	12071	12.42	1.03	3.10
1987	4	12435	9.87	0.79	2.47
1988	5	16030	9.36	0.58	1.87
1989	5	14565	10.37	0.75	2.07
1990	5	14665	8.96	0.61	1.80
1991	6	13184	9.79	0.74	1.63
1992	6	11552	10.6	0.92	1.77

Table 3: Occupational dose distributions in NPP „Kozloduy" for 1987 - 1992 period

Department	Number of personal annual doses	Distribution of the personal doses, % Dose intervals, mSv				
		J 4	4 - 12	12 - 30	30 - 50	50 - 70
Head staff	102	90.2	4.9	4.9	-	-
Reactor repearing operation	2699	40.6	22.9	20.9	14.9	0.7
Radiation protection and dosimetry	978	84.2	12.4	3.4	-	-
Techn. measur. and automat.	984	93.6	5.7	0.7	-	-
Menagment of nuclear safety system	323	93.8	5.9	0.3	-	-
Reactor's operators	417	86.8	12.5	0.7	-	-
Research department	232	71.6	24.1	3.9	-	-
Electricity maintenance	933	88.0	11.4	0.6	-	-
Chemical analyses	409	90.5	8.6	0.7	0.2	-
Technolog repairing deparrtment	206	69.9	23.3	6.8	-	-
Centralized electroequipment repair	588	63.1	20.4	16.0	0.5	-
Specialized building repair	613	46.5	19.1	29.9	4.6	-
Turbine maintenance	48	83.3	6.3	10.4	-	-
Russian specialists	497	72.8	20.1	6.0	0.6	0.4
Radioactive waste processing	102	67.6	26.5	5.9	-	-
Other organizations	2067	70.4	18.2	8.8	2.7	-
TOTAL	11198	68.6%	16.6%	10.2%	4.4%	0.2%

BG9700056

ON THE PROBLEM OF THE RADIOPROTECTIVE FOODS

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After the accident in the Chernobyl NPP the searches in the comperatively new radiation protection field - creation of radioprotective foods - are considerably intensified in the country. The elaboration of such products is justified from a scientific as well as from a practical point of view for use of the population in case of radiation accident situations. The means of chemical radioprotection or drugs for decorporation of the radioactive substances intaken in the human organism cannot be applied for series of reasons on large population contingents. Most of these drugs can be used only on medical advice. Studies on chelating agents application by drinking water are still in initial phase/10/. In case of radioprotective foods,it is aimed the elaboration on a large scale usage foodstuffs containing harmless components manifesting one or other antiradiation qualities. It is envisaged including in some standard foodstuffs biological active products of natural origin for which there are data for direct or indirect antiradiation effects - vegetable fibres, vitamins, carotins, natural mineral waters, polyphenols, aminoacids, micro and macro elements, etc., /3,5/. These are substances which action mechanism influence the nonspecific immune resistance or decrease the accumulation of radioactive materials in the organism - principally by restricting the ingestion in the gastrointestinal tract. Antiradiation foods on the basis of milk, bread, meat, fish, brewery products, foods on fruit and vegetable basis, etc., were created on this principle in our country for a period of 5-6 years/1,3,6,7,8,9,11/.

In the survey of the proposed 'antiradiation foods' the detailed elaborations of technologies for production, supporting the composition, stability in safe-keeping, creation of possibilities for varying the composition according to the available raw materials,etc.,make impression.These elaborations are on the background of the relatively limited biological studies for proving the radioprotective properties of the proposed foods.The biological researches are more frequently restricted to:

1. Demonstration of the decorporation possibilities(rather restricting the resorption in the gastrointestinal tract) towards the radioactive ceasium and strontium in case of small laboratory animals in the conditions of acute experiments. These tests don't present convincing data for extrapolation to man.

2. Demonstration of their favorable influence in the case of some ionizing radiation deterministic effects on the hemopoetic and the immune systems. The experiments are carried out on small laboratory animals. These data extrapolation to man is too much uncertain.

3. Demonstration of the harmlessness of the products in case of people's relatively prolonged use.

4. Application to patients with malignant processes subjected to irradiation treatment. These results can be treated as lack of aggravating the patients' conditions than as establishing a positive effect.

Experiments for demonstrating the influence of the created products on the ionizing radiation stochastic effects are not carried out. The general conclusion of the up to date medico biological studies is for the properties of these foods one can judge indirectly only by the compounds including in them. There are not reasons to attribute them the specific antiradiation quality.

The antiradiation foods elaboration has a basic importance only of the point of view of the population accident radiation protection. Their usage in a normal situation is not grounded.In the country there are not regions or enterprises where the application of these foods has practical importance.

In the case of the personnel of the NPP "Kozlodui" their application is too nonprospective of the point of view of the internal and external irradiation:

1. In case of decorporation - the internal radiation doses received from the personnel represent only 14-24% from the total irradiation, while in the cases with the highest content of radionuclides the found activity is under 10% of the limits of the annual intakes /2/. The intake is principally by inhalation. So measures must be directed to protect the respiratory tract.

2. Action against the external irradiation - the data received up to date give reasons, even with stipulation, to assume a positive action of the proposed foods for some deterministic effects. These effects can be avoided following the norms for radiation safety /4/. In this case the protection from the stochastic effects is important. There are not such studies with the proposed foods.

It is seen both for the internal and external irradiation that the nature of the necessary protection must be directed otherwise. The application of the proposed antiradiation foods would cause more damage creating a wrong notion for protection.

The elaborated foods can find their application because of their composition, not only in the case of radiation accident,but in the treatment of some diseases, rehabilitation, prophylactic medicine, etc.

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BG9700057

WHOLE-BODY DETECTOR CALIBRATING WITH A MODULAR PHANTOM*

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INTRODUCTION

On purpose to evaluate the human body activity, the gamma-ray spectrometers calibration is done with the help of the so called phantoms, which represent variable human body modules: from tubes (cylinders) (their size being close to the standard human) to the realistic phantoms (representing plastic toys with a real human shape, containing all of the significant organs).

The phantoms of the first kind, roughly imitate the human body, with an appropriate geometric form, in which either the homogeneous activity is represented with a limited number of standard point sources, put in suitable places, or the volume is filled with standard radioactive solution.

The shortcomings of these phantoms are: the human body shape is rather approximate; bodies of different size and geometries are hard to reproduce; no activity can be imitated in the critic organs because of their unified volume. On the other hand, their price is quite low and they can easily be reproduced in a laboratory.

The realastic phantoms precisely represent the human's shape and contain the most important organs. Every organ has a separate volume and can be independently filled. The density and Zeff of the organs are very close to the real values. This way, both homogeneous and certain organs activity can be reproduced. Their most significant shortcomings are their high price and the impossibility of reproducing bodies of different size, as only standard human size bodies are manifactured.

The set of module phantoms that we propose, occupies an intermediate position. Its advantages are:

- it allows the moulding of human bodies of a shape, close to the real and of an arbitrary mass (to 100 kg);
- different geometries can be reproduced (a lying man, a sitting man, etc.);
- the detector's background for the chosen geometries can be evaluated;
- it allows the moulding of a human body with activity concentrated in a certain organ (lungs, thyroid gland, liver, etc.).

1. MAKING OF MODULE PHANTOMS FOR SEMICONDUCTORS CALIBRATING ON PURPOSE TO EVALUATE THE ACTIVITY, INCORPORATED IN THE HUMAN BODY

The phantoms of this kind are made on the principle of modules: a satisfying number of modules of the right shape and mass are prepared. Thus human bodies (phantoms) of different size in different geometries can be moulded. The availability of small enough modules enables the human critic organs moulding. In case the modules contain standard activity, the corresponding phantoms can be used as standard samples for the gamma-ray spectrometric system calibrating. With the help of non-containing activity modules, "zero"-phantoms for the detector background calibrating can be moulded. The availability of enough modules (with and without activity) enables the system calibrating in case the activity is homogeneously distributed in the human body, as well as in case it is concentrated in a certain critical organ. In the first case, the whole phantom is constructed from containing standard activity modules, in the second the modules, imitating a critic organ and containing standard activity, are placed in the "zero"-phantom.

Common considerations make it clear that the smaller the module's mass (size) is, the more precisely bodies of an arbitrary size would be made. The small moduls enable

* Work supported by NPP-Kozloduy

the forming of an arbitrary geometry. Besides, the smaller the module is, the less the influence of the eventual unhomogenity of the activity within the framework of the module would be.

Having this in mind, it is expedient to make the mass of the homogeneous activity reproducing module 0.5 kg. Thus, a small phantom (40 kg) will be made from 80 modules, and the largest (100 kg) - from 200 modules. Having so many modules, moulding the phantom, it is important that all of them contain THE SAME radioactivity.

The size of the critical organs reproducing module, was determined by the size of the kidney which, according to [1], is 10-12 x 5-6 x 3-4 cm.

The devices for preparing the modules and the techology of their using are described in detail in [2]. With them, the following modules were made:

- "zero" - phantom modules, each one with 0.5 kg mass and size 20 x 14 x 2 cm, 180 pieces, not containing radioactivity;
- modules, containing standard Eu-152 and Am-241 radioactivity, each one with 0.5 kg mass and size 20 x 14 x 2cm, 180 pieces, designed for homogenious radioactivity imitating;
- modules, containing standard Eu-152 and Am-241 radioactivity, each one with 0.16 kg mass and size 11 x 9 x 0.5 cm, 20 pieces, designed for "critical" organs moulding.

The radioactivity containing modules are reliably packed. The outside package is an envelope, made of impregnated clothe.

2. CALIBRATING OF THE SYSTEM FOR THE INCORPORATED IN THE HUMAN BODY ACTIVITY EVALUATING

2.1. A brief description of the EP2 whole-bodi system

The detector shielding belongs to the so called "shadow" shields. It is a copy of the EP1 system and was made with the assistance of NCRRP. Its construction is shown on fig.1 and 2.The walls are made of lead plates with total width 100 mm, the inner sheath of copper. The section of the opening in which the lying man moves, is 800 x 620 mm. The side walls are trapezium - shaped with 2030 mm length. A wheelbarrow with a bed on it moves along rails on the bottom of the shielding. A low-voltage electromotor is directly affixed to one of the wheels. Through an appropriate transmission it provides speed 2 mm/s. A HpGe detector with relative efficiency 19.9% is used. It is affixed to the top plate of the shielding and is not additionally enclosed by lead. The whole construction is installed upon a ferro - concrete foundation, 4800 mm long and 300 mm thick. After we accomplished a number of preliminary measurements, in answer to our suggestion, some changes were made in the system. The detector was lifted up so that it did not project out of the top of the shielding, and the opening through which it entered the shielding, was additionally protected. That raised the shielding coefficient from 11 to 30. A mechanism lifting the bed, was also made. Thus the man/detector distance can be kept the same, despite the size of the object. Besides, the drawing of the bed to the detector, raises their radiated gamma-quanta registration efficiency. A construction, allowing affixing of collimators for testing the possibility of locating the activity in the human body through scanning, was also built.

2.2. Human bodies of different size moulding by the modules adequate arrangement

People's antropological size was evaluated according to [1,3]. Following the data from [3], the weight/height relation curve was drawn.

The size of the first two phantoms was conformed to this curve. They imitate human bodies with the following proportions: 40 kg weight/height 150 cm and 60 kg weight/height 170 cm. As this curve reaches saturation when the height is short, the last two phantoms imitate bodies of taller stature - 75 kg weight, height 185 cm and 95 kg weight, height 185 cm. It should be considered that the weight of the phantoms is not equal to the respective human's weight. As the gamma-ray radionuclides are mainly concentrated in the soft tissues (not in the bones), the mass of the skeleton (which is about 15 % of the total mass) is substracted from the human's weight.

The phantoms used for calibrating the gamma-spectroscopic system, had the following final size:

- 35 kg weight, height 150 cm - corresponding human's weight 40 kg;
- 50 kg weight, height 170 cm - corresponding human's weight 60 kg;
- 65 kg weight, height 185 cm - corresponding human's weight 75 kg;
- 80 kg weight, height 185 cm - corresponding human's weight 95 kg.

The ways of arranging the 35, 50 and 65 kg phantoms, are shown on fig. 3. The main part of the 35 kg weight phantom, is made of 4 layers of modules (in thickness), the 50 kg weight one - from 5 layers of modules and the 65 kg one - from 6 layers of modules. The 80 kg weight module was made by adding another layer of modules (mainly in the trunk) to the 65 kg one. This way of arranging the modules facilitates the phantoms of different shape reproducing.

2.3. Calibration of the system

After some preliminary measurements, orientated to establishing the optimum measuring conditions, with the help of these phantoms, the calibration curves (the relation to the gamma-quanta energy) for certain conditions, were drawn.

The absolute efficiency values for some energies of concrete radionuclides, which are expected to be identified in the NPP staff, are given in table 1. The results in the first 4 columns refer to the measurings in the following conditions: standard motion of the bed (2 mm/s), homogeneously distributed activity and distance between the top surface of the phantom (the man measured) to the detector, equal to 6 cm.

Table 1

E, keV/	Absolute efficiency, $x10^{-4}$				
nuclide	35 kg	50 kg	65 kg	80 kg	lungs
604/Cs-134	1.60	1.42	1.26	0.870	4.03
657/Ag-110	1.51	1.33	1.19	0.823	3.82
661/Cs-137	1.50	1.33	1.19	0.820	3.80
795/Cs-134	1.32	1.16	1.05	0.726	3.27
834/Mn-54	1.28	1.13	1.02	0.703	3.20
884/Ag-110	1.23	1.08	0.987	0.677	3.08
1173/Co-60	1.02	0.886	0.828	0.563	2.62
1332/Co-60	0.940	0.810	0.767	0.519	2.33

With the help of the radioactivity non-containing modules and a part of the designed for imitating "critical" organs modules, calibrations for activity, concentrated in the lungs, were also accomplished. 14 modules of the kind were placed in a phantom in a way that they would aproximately imitate the shape, size and location of the lungs in the human body. The distance between the top surface of the phantom to the detector is 6 cm again. The respective absolute efficiency values, evaluated through measuring without motion and non-homogeneous distribution of the activity, are given in the last column of table 1. It should be considered that this calibration can be used only if after a double measuring [4], it is ascertained that the incorporated activity is concentrated in the lungs.

2.4. Relation of the efficiency to the mass (size) of the phantom

To study the relation of the efficiency to the mass of the measured object, it is needed to keep the distance between the top surface of the phantom to the detector, the same for all of the phantoms. On that purpose, a mechanism moving the bed in the vertical direction to an appropriate position, was installed so that the aforesaid term would be realized. Thus the influence of the source-detector distance factor would be reduced, which allows a clearer delineation of the curve, showing the relation of the efficiency to the mass of the meausured human.

The results obtained through the measuring of different phantoms, the distance to the detector being 6 cm, are displayed on fig. 4. Having in mind that the efficiency depends not only on the mass, but on the geometric size of the measured object as well, the relation of the efficiency to the mass/size [kg/cm] ratio is shown on fig. 4.

This relation is obviously a complex function. The drawing of an analytic expression requires the introducing of a number of limiting and simplifying terms, which are hard to realize and control during practice measurings. This necessitates a polynomial approximation of the experimentally obtained in certain conditions efficiency values. The experimentally obtained efficiency values in different conditions (different size of the phantoms) for some concrete gamma-quanta energies, are shown on fig. 4. The small number of the experimental points in this relation, added to the inevitable random and systematic uncertainties, does not allow the using of a relation more complex than a linear one, for their extrapolation. The using of a polynomial of a higher degree would lead to unsteady solutions, which seem better to describe the experimental points, but in fact the polynomial coefficients would be more influenced by the uncertainties than by the real characteristics of the measured object.

That is why a linear relation was used - using the least squares method, a straight line was drawn through the experimental points. In table 2, we have given the coefficients a and b of the respective line $f(m/l) = a.m/l + b$, where m is the mass of the phantoms in kg and l - its size in cm.

Table 2

	with motion	
E, keV	a x10^{-4}	b x10^{-4}
122	- 7.09	+ 4.86
344	- 5.25	+ 3.59
779	- 2.84	+ 2.00
1408	- 1.85	+ 1.33

3. ANALYSIS OF THE POSSIBILITY OF SCANNING THE HUMAN BODY ON PURPOSE TO LOCATE THE ACTIVITY, INCORPORATED IN IT

An answer to the question where the incorporated activity is concentrated, can be obtained if the total counting rate is registered as a function of the position of the detector toward the body, during the human body's motion. That is why the total number of pulses from the outlet of the spectrometric amplifier, were registered not only by the analysator but also by the 20046 radiometer, which was exploited as a counter with measure time 20 s.

3.1. Profiles of the activity in the human body without using collimators to the detector

3.1.1. Homogeneously distributed activity profiles

Human bodies weighing 35, 50 and 65 kg and with homogeneously distributed activity in them, were contiguously moulded on the bed, with the help of the modules, designed for homogeneous activity. The distance between the chest and and the detector was determined 6 cm. The bed was moving in a standard way: range of shifting - 175 cm, one-direction shifting time - 870 c 15 s, that is the motion speed was 2 mm/s. The readings of the counter were recorded only in one direction (relative motion of the detector from the head to the legs). The results are displayed on fig. 5. On the axis of abscissae, the points (over the phantom) in which the reading was done, have been projected, that is they show the position of the detector above the phantom at the moment of reading. On the axis of ordinates, the sums of pulses for the chosen measuring time - 20 s, have been projected. The figure shows that:

- the sum of pulses is proportional to the the size of the phantom, that is to its total activity;
- the following areas form in the profiles: abrupt rising of the sum of pulses - when the detector enters the trunk, "plateau" - when the detector moves over the head or the trunk, and gradual reducing of the sum of pulses - when the detector moves over the legs, toward the feet.

3.1.2. Profile of the activity in a critical organ

With the help of the "zero"- phantoms, a standard human body was moulded on the bed. With the critical organs modules, the following cases were imitated in it:

- activity in a critic organ (lungs) only;
- activity in critic organ - lungs and kidneys;
- activity in critic organs - lungs and kidneys, with additionally "externally polluted" hair and hands.

The result profiles in those three cases scanning, are displayed on fig. 5. It shows that despite the lack of a collimator, the presence of activity in (a) critic organ(s) forms maximums. Besides, the energy resolution of an uncollimated detector is rather low. The figure also shows that the activity in a critic organ (lungs) only, forms a maximum with center over the lungs (curve 1). The adding of kidneys, hands and hair just expands the maximum in the respective direction (curves 2 and 3).

The profiles would be more distinguished, if a collimator is installed under the detector, so that the detector can "see" only a certain sector of the human body. It is clear that the narrower the slit is, the higher the energy resolution of the scanning system and the more precise the locating of the incorporated activity will be. At the same time, however, the efficiency of the system (the counting rate) will be reduced. That is why, a suiting compromise between a high energy resolution of the scanning system (a narrow slit) and a high enough efficiency (high counting rate).

In answer to our suggestion, in the machine department of EP2 a simple elbow construction enabling the lead collimator (10 cm thick and of an arbitrary width) installing under the detector, was built up.

Profiles of the activity in the human body with the detector collimating.

The described in 3.1. cases of activity in the lungs, kidneys, and "externally polluted" hair and hands, were reproduced.

A collimating slit (5 cm thick and 5 cm wide) was installed under the detector. The respective profiles during the motion of the bed in one direction - from the head to the feet, were taken. The profiles obtained through the continuous scanning of these models, are shown on fig. 7. The difference between it and fig. 6 (profiles in the same cases but without collimating the detector), is obvious. The profiles obtained with the help of a collimated detector, allow a much more precise locating of the incorporated activity in the human body. When the activity is concentrated in the lungs only, a narrow maximum with center over them, is formed. The adding of activity in the kidneys, forms a clear plateau on the profile curve. The "externally polluted" hands and hair, form independent maximums in the respective places. All that shows the indubitable advantage of using a collimator when scanning the human body on purpose to locate the incorporated activity in it.

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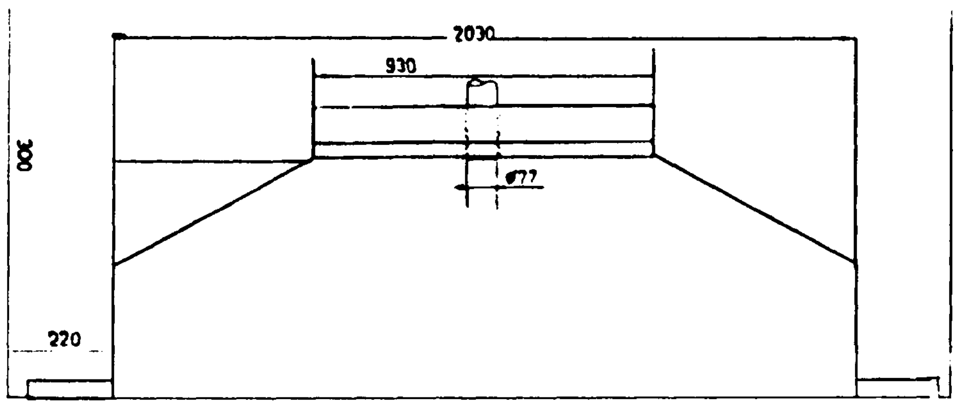


fig. 1

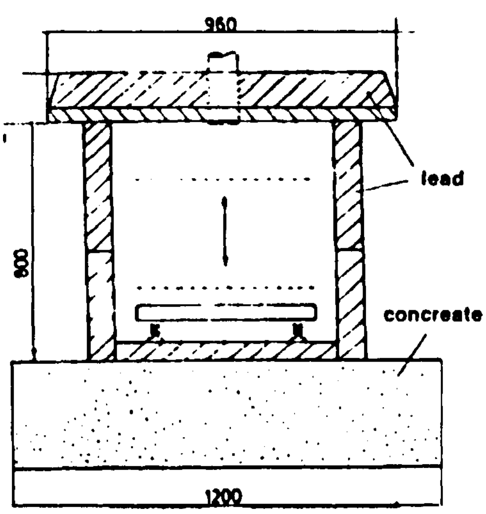


fig. 2

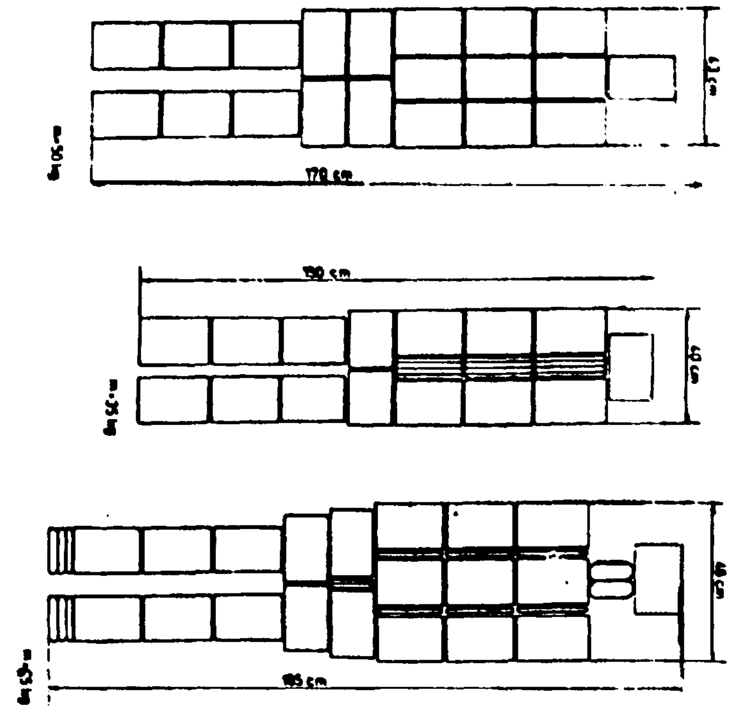
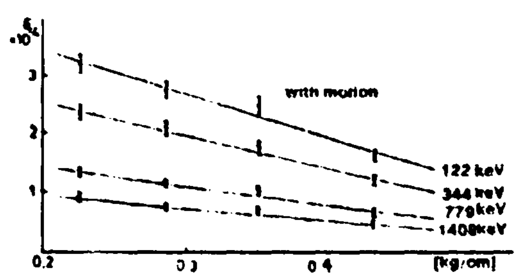


fig. 3



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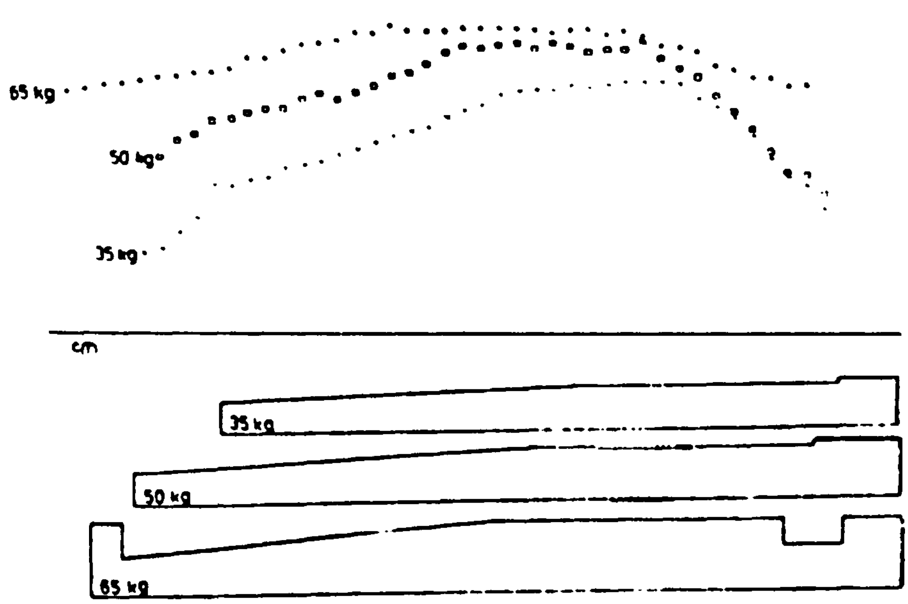

fig. 5

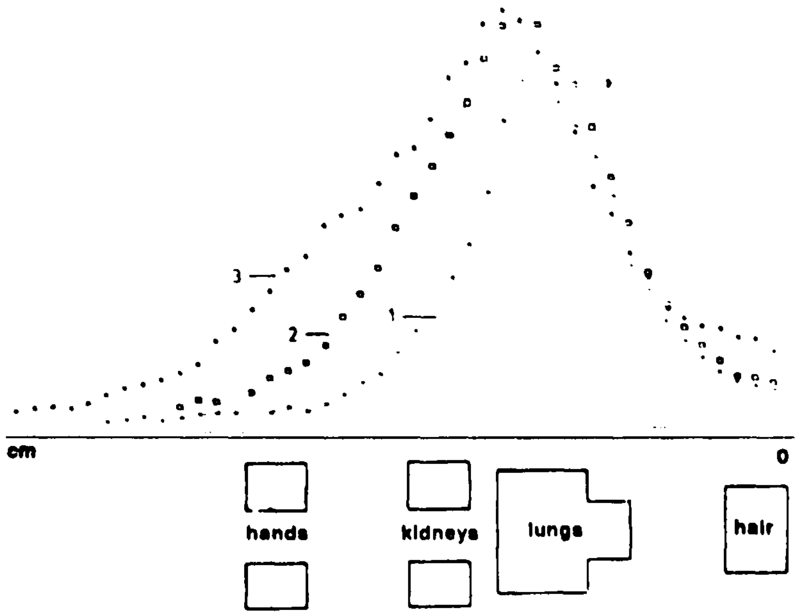
3
2
1
cm
0
hands
kidneys
lungs
hair

fig. 6

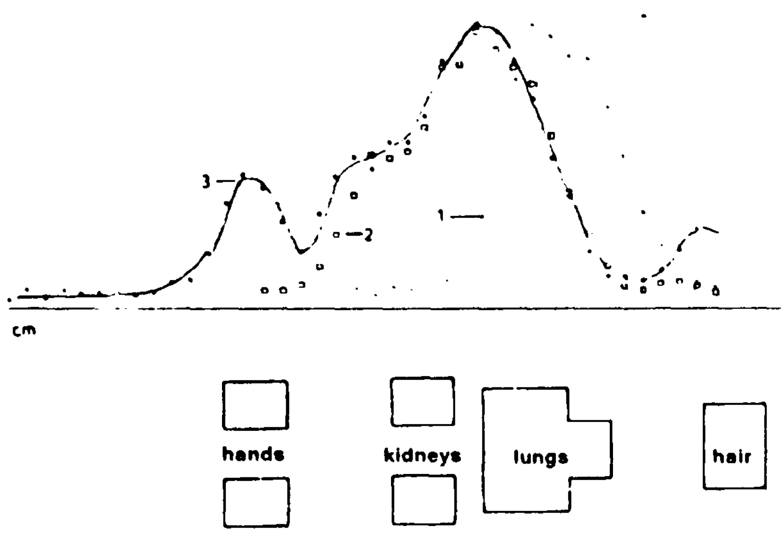
3
2
1
cm
hands
kidneys
lungs
hair

fig. 7

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ASSESSMENT OF THE ACTIVITY INCORPORATED IN THE HUMAN BODY, BY MEANS OF A HpGe - DETECTOR*

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INTRODUCTION

The first attempts to evaluate the activity incorporated in the human body, refer to Schlundt's [1,3], Flinn's [2], Evans' [4] works. Already in the 30ies, detectors were calibrated with the help of a phantom and attention was paid to the measurement configuration influance, the selfabsorption of the body, the reducing of the background by the measured object.

There is not a standard measurement configuration for the wholebody activity yet. Some of the most used geometries, are a "chair" and a "bed". The "chair" geometry was introduced by Marinelli in the middle of the 50ies. The measured man is sitting on a chair, the angle between the chair-bottom and the back being 90ш. An advantage of this geometry is the approximately equal distance from the different parts of the body (except the feet) to the detector. This provides comparatively precise results, undependent on the activity distribution in the body. Another variant of the same geometry was offered by Palmer [5]. (Dr. V. Bosevsky uses it [6]). The sitting man folds the detector, which is placed on his lap. This raises the efficiency but can also cause significant errors because of the unhomogenity of the activity distribution. A third variant is to place the detector under the chair-bottom, but the uncertainties are more significant than in case the detector is put above the measured man [7]. (This variant was used by Mandjukov [8] after the Chernobile failure).

The "bed" geometry was described by Wade [9], Owen [10] and Rundo [11].The measured man is lying, and the detectors (one or more) are placed over and/or under the bed. An important advantage is that it enables the scanning [12,13].

The radionuclides comparatively low content in the human body makes their precise monitoring in a natural gamma-background rather difficult. Besides, the shielding effect of the human body increases the uncertainty when evaluating his self-activity. Building a shield around the measured object/detector system reduces the uncertainty, caused by the above-mentioned problems. The most used shields are the shielding monitoring room and "shadow" ones. The advantages of the shielding monitoring room, are: high protective coefficient, measurement configuration free choice, a long distance between the detector and the shielding (respectively, the scattered radiation registered by the detector, would be less), the claustrophobian problems are minimum. However, their high price is a serious shortcoming.

In many cases of the whole-body counting, a partial (shadow) shield is used - it covers the detector and just a part of the human body (usually the one to which the detector is turned). The shadow shielding type is directly dependent on the measurement configuration.

The results represented in this work, were obtained during the measurements, carried out in a laboratory for evaluating the activity incorporated in the NPP-Kozloduy (Part 2) staff. The measurement configuration is a bed, the shielding - a shadow one. The system (configuration and shielding) is the same as the Part 1 one and was built with the help of NCRRP specialists. The detector used is a p - type Ge semiconductor detector (PGT - Germany) with 19,9 % relative efficiency and resolution 1,9 keV for 1332 keV.

1. MEASUREMENT CONDITIONS CHOICE

The perfect measurement configuration presumes:
- high gamma-quanta registration efficiency;

* Work supported by NPP-Kozloduy

- minimum uncertainties, caused by the alteration of some of the object's parameters (size, activity distribution, etc.).

In the real measuring a satisfactory compromise between the two, has to be assumed, dependent on the aim of the measurement. In case of a failure, e.g., the high efficiency is more important and for ordinary measurements - the uncertainties minimizing.

During 1992 - 1993, in a NPP-Kozloduy (Part 2) laboratory, a series of researches were carried out, aimed at defining the best condiotions for the NPP-Kozloduy staff's whole-body counting (in ordinary work conditions). On that purpose, module phantoms were used [14], enabling the imitation of the activity homogeneous and unhomogeneous distribution in the human body. The system allows measurements both with the bed moving under the detector (v = 2 mm/s) and with a steady detector (it is fixed above a certain critic organ). Plenty of measurements of phantoms of different size and in different positions were completed. Modules, containing standard activity, placed in a "zero" - phantom, were used for imitating activity, concentrated in a critic organ. The detector was fixed above the chest in the measurements without motion of the bed.

The following ralations were studied:

1.1. Relation of the efficiency to the measurement conditions (with/without motion)

The results from the measuring of a phantom (50 kg), imitating homogeneously distributed activity, and a phantom, imitating activity concentrated in the lungs, are shown in table 1. Measurements with different distance to the detector, were made (R - the distance between the bed surface and the detector). The absolute efficiency and its relative variation for the gamma-quanta different energies, are given in the table.

Table 1.

E keV	homogen. distr. activity			activity in the "lungs"		
	$e_{h,m}$ $\times 10^{-4}$	$e_{h,s}$ $\times 10^{-4}$	e_h [%]	$e_{c,m}$ $\times 10^{-4}$	$e_{c,s}$ $\times 10^{-4}$	e_c [times]
			R = 42 cm			
60	0.286	0.385	35%	0.470	1.14	2.4
122	1.450	2.010	39%	2.620	5.77	2.2
344	1.000	1.410	41%	1.700	3.93	2.3
779	0.556	0.783	41%	0.855	2.01	2.3
1408	0.384	0.534	39%	0.566	1.33	2.3
			R = 24 cm			
60	0.381	0.492	29%	0.537	2.08	3.9
122	2.57	3.57	39%	3.31	11.45	3.5
344	1.81	2.58	42%	2.34	7.21	3.1
779	1.02	1.44	41%	1.40	4.06	2.9
1408	0.706	1.01	43%	0.894	2.58	2.9

$e_{h,m}$ - homogen. distr. activity measuring, with motion
$e_{h,s}$ - homogen. distr. activity measuring, without motion
$e_{c,m}$ - "lungs" activity measuring, with motion
$e_{c,s}$ - "lungs" activity measuring, without motion

$e_h = (e_{h,s} - e_{h,m})/e_{h,m}$

$e_c = e_{c,s}/e_{c,m}$

Common considerations make it clear that the gamma-quanta registration efficiency would be higher when measuring without motion than with motion. Table 1 shows that this increase is 30-40% for homogeneously distributed activity, and 2-3 times (dependent on the distance to the detector) when the activity is concentrated in the lungs.

The relation of the efficiency variation (respectively, the counting rate in the full energy peak) to the activity distribution, when changing the measurement conditions, can be used for determinating whether the activity is homogeneously distributed, or concentrated in a certain critical organ.

1.2. Relation of the efficiency to the activity distributon

The whole-body counting systems are usually calibrated with phantoms, containing homogeneously distributed in the body activity. In the practise, however, it is very probable for the activity to be concentrated in certain organs, the so called "critic organs". This discrepancy may lead to serious errors when evaluating the dose absorbed.

On purpose to assess the mistakes of this kind, plenty of measurements were made - with and without motion, for phantoms homogeneously and unhomogeneously distributed activity. The evaluations, displayed in table 2, are based on the measurement, described in 1.1.

Table 2.

E keV	measuring with motion			measuring without motion		
	$e_{h,m}$	$e_{c,m}$	e_m	$e_{h,s}$	$e_{c,s}$	e_s
	$\times 10^{-4}$		[%]	$\times 10^{-4}$		[%]
			R = 24 cm			
60	0.381	0.537	41%	0.492	2.08	320%
122	2.57	3.31	29%	3.57	11.45	220%
344	1.81	2.34	29%	2.58	7.21	180%
779	1.02	1.40	37%	1.44	4.06	180%
1408	0.706	0.894	27%	1.01	2.58	155%

$e_m = (e_{c,m} - e_{h,m})/e_{h,m}$

$e_s = (e_{c,s} - e_{h,s})/e_{h,s}$

The table shows that the uncertainties, caused by an incorrect calibration, range to 30-40 % in the measurements with motion, while in measurements without motion (the detector is fixed above the chest), they can reach 150-300 %.

Having in mind that in practice it cannot be preliminarily known whether the activity is homogeneously distributed, or concentrated in a critic organ, the standard motion of the bed seems an optimum measurement condition.

The relation of the efficiency to the activity distribution in "thickness", was also studied (with standard motion of the bed). It was ascertained that if the position of the measured object does not change in the course of measurement, the difference between the efficiency for homogeneously distributed activity, and activity concentrated in the bottom or upper layer, can reach 30-80 % (dependent on the gamma-quanta energy). But if the position of the layers changes during the same measurement (e.g. if the object is measured when lying prone and on his back) the uncertainty can be reduced to 5-15% .

1.3. Relation of the efficiency to the weight (size) of the object measured and to the distance to the detector

On purpose to study the relation of the efficiency to the mass, phantoms of different size were moulded:

- a phantom with 35 kg weight and 150 cm long (m/l = 0.23);
- a phantom with 50 kg weight and 170 cm long (m/l = 0.29);
- a phantom with 65 kg weight and 185 cm long (m/l = 0.35);
- a phantom with 80 kg weight and 185 cm long (m/l = 0.43).

Evidently, when the distancc between the bed and the detector remains the same, the alteration of the phantom's size, leads to the phantom/detector distance varying. Consequently, the relation of the efficiency to the phantom's mass is not clearly distinguished. Fixing a mechanism lifting the bed, allows to keep the distance between any single object and the detector, the same The availability of such a mechanism as well as of the above described phantoms, let us study the relation of the efficiency both to the size of the measured object (the distance between it and the detector being fixed), and to the distance to the detector (for the fixed size of the object). The experiment showed that the relations are more strong in measurements without motion. So, in case of a difference between the measured object's size and the phantom's size (used in the calibration curve), the uncertainties would be less significant when measuring with standard motion of the bed.

Having in mind that the optimum measurement configuration should provide not only minimum systematic uncertainties, but a comparatively high gamma-quanta (radiated by the human body) registration efficiency, it is purposeful to keep the distance to the detector comaparatively small during the measurements.

1.4. The measured object's shield effect influence

When measuring in a low-level shield, it is necessary to have in mind the shield effect of the non-radioactive part of the measured object.

The results, obtained in measuring the background both in an empty shield and with a "zero"-phantom, are given in table 3. In the first measurements, a part of the detector remains outside the shield, and so the shielding coefficient is really low (k = 1). The "zero"-phantom is placed about 20 cm away from the detector in that case. Later the detector was taken inside the shield, so that its forehead is at the same level as the top of the shield. This raised the shielding coefficient (k = 30). In that case the "zero"-phantom is placed 6 cm from the detector. The counting rate [imp/s x 10^{-2}] for some typical lines of the gamma-background, are given in the table.

Table 3

E keV	k = 11, r = 20 cm		k= 30, r = 6 cm	
	empty shield	"zero" phantom	empty shield	"zero" phantom
352	1.98±0.16	1.98±0.12	0.726±0.174	0.247±0.074
609	1.65±0.10	2.13±0.08	0.439±0.066	0.234±0.042
1461	6.85±0.14	6.69±0.13	2.15 ±0.09	1.38 ±0.07

It can be seen that in the first case, the counting rate with a "zero"-phantom is statistically equal to the counting rate in an empty shielding. In the other case, the shield effect of the "zero"-phantom, is more distinguished.

Having in mind that the low activities evaluations uncertainties are much dependent on the background corrections, it is necessary to evaluate the shield effect of the measured object, for every single change in the measuring conditions.

The following measuring conditions were established, according to the above described measurements results:

-standard measurement with motion

The measured man lies on the bed, which is lifted or lowered so that the distance between the chest and the detector is 6 cm. A measuring cycle includes horizontal moving of the bed in both directions; the man passes under the detector twice: once lying prone, and once lying on his back. The availability of preliminary calibrations (implemented with the help of phantoms with homogeneously distributed activity [14]), enables to evaluate activity incorporated in the human body, using the spectrum already obtained. The data from that measurement is not enough to locate where the registrated activity is concentrated, the quantity evaluation is precise enough (30-40 %), not depending on the activity distribution. The results obtained are in [Bq/body]. Turning them to specific activity units [Bq/kg] is incorrect when it is not ascertained whether the activity measured is homogeneously distributed or not.

- additional measurement without motion

This measurement is valuable only as a secondary (following the standard) one, on purpose to find out whether the incorporated activity, is concentrated in the lungs. The measured man is placed on the bed, the detector is fixed above the chest (6 cm from it). The high energy resolution of the HpGe - detector allows to evaluate the difference of the counting rates for the two measurements, for every single gamma-line (respectively, nuclide). According to the evaluations from 1.2., when the difference is about 30-40 %, it is assumed that the activity is homogeneously distributed. Anyway, if the counting rates differ more than 2,5 times, the activity registrated is concentrated in the lungs.

2. EVALUATION OF THE INCORPORATED IN THE HUMAN BODY RADIONUCLIDES ACTIVITY

The methods for determinating the nuclides content and the activity of the artificial gamma-radiating nuclides, incorporated in the NPP-Kozloduy staff, Part 2, in normal work conditions, are fully described in [15].

2.1. Determinating the detectability limits

To determinate the detectability limits, it is necessary to evaluate the respective "critical" levels and the detection levels.

The "critical" levels R(E) are used only as solution levels - whether the counting rate in a certain interval of a man's spectrum is statistically different from the counting rate in the respective energy interval of the background spectrum [16]

$$R(E) = k_{\alpha} * \sigma_{b}(E) * \sqrt{1+\tau}$$

where: k_{α} is the coefficient for one-sided confidence level;

σ_b (E) is the standard deviation of the counting rate in a certain energy interval of the background spectrum;

τ = T/t, T is the background measurement time, t - the person's measurement time.

In contrast to [16], in the present work the detection levels D(E) are defined as the maximum true (for a certain confidence interval) counting rate, that could be expected when the registrated "pure" counting rate is equal to the respective "critical" level. Then:

$$D(E) = R(E) + \sqrt{\left[R(E) + k_{\alpha}^{2} / t\right] * R(E)}$$

The detection levels D(E) allow to evaluate the top limit of the activities that cannot be detected in those measurement conditions, even if they are present in the human body ("undetectable activity - UDA").

$$UDA(E) = \frac{D(E)}{C(E)},$$

where C(E) is the calibration coefficient for the certain meaurement conditions.

This way the detection limits for concrete radionuclides, can be estimated. Unlike the low-background gamma-ray spectroscopy of nature samples, when measuring whole-body activity, it is possible to evaluate the detection levels on the basis of a background spectrum. For that purpose it is necessary to add KMnO4 (or KCl) with correspondent to the K-40 content in the human body activity, to the "zero"-phantom. (Anyway, we must have in mind that in case there are high-energy gamma-quanta radiating artificial radionuclides in the human body, the detection levels evaluating for lower energies, should be based on the spectrum, obtained through the man's measurement). The identity of the spectrum of a man in whose body there are no artificial radionuclides, and the "zero"-phantom's spectrum, allows the usage of a "critical" level, determinated on the basis of the whole energy interval (50 - 2000 keV). This significantly shortens the time for processing all the spectra, obtained by measuring a large number of people.

In case the difference between the counting rates of the man's spectrum and the background spectrum is less than or equal to the respective "critical" level (for a certain energy interval), it is assumed that even if there are incorporated artificial radionuclides in the human body, their activity is less than the detection limits.

2.2. Evaluating the activity in standard measurement conditions

If the counting rate in a certain energy interval of the measured man's spectrum is statistically different from (higher than) the respective counting rate in the background spectrum, it is assumed that there is a positive effect of the man's measuring and the activity [Bq/body] is evaluated:

$$A(E) = \frac{q(E)}{C(E)} \pm \delta A$$

where q(E) is the counting rate in the full energy peak net area (with energy E), registrated in a standard measurement of a man [imp/s];

C(E) is the calibration coefficient [imp/s.Bq];

δA is the result's total uncertainty [%].

The confidence interval of the total uncertainty, is determined by:

$$\delta A = K_{\beta} * \sqrt{\sum_i (\sigma_i^2) + \frac{1}{3} * \sum_i (\theta_i^2)}$$

where σ_i and θ_i are the random and systematic uncertainties in [%],

K_{β} is the two-sided confidence interval coefficient.

The random uncertainty is determined mainly by the standard deviation counting rate q(E).

The systematic uncertainty is represented with:

$$\sum_i (\theta_i^2) = \theta_e^2 + \theta_r^2 + \theta_m^2 + \theta_{a1}^2 + \theta_{a2}^2$$

where:

θ_e^2 represents the uncertainty, relating to the calibration curve drawing. In the concrete case θ_e is estimated to 6 % ;

θ_r^2 represents the uncertainty, relating to the distance r [cm] (between the object measured and the detector) reproducing. For r = 6 cm, uncertainties within 1 cm, ead to θ_r = 7 % ;

θ_m^2 represents the uncertainty, relating to the difference between the "man's" size and the standard model's size, used for the system's calibration. The availability of a modular "phantom" enables the calibrating with a variety of phantoms of different size, so that the uncertainty θ_m is reduced to 10 % ;

$\theta^2_{a1,2}$ represent the uncertainties, relating to the activity distribution in the human body. Having in mind that the calibration coefficient C(E) was determined on the basis of a homogeneous distribution, θ^2_{a1} is connected with the uncertainties that appear in the measurement of activity, concentrated in a "critical organ". θ_{a1} reads the horizontal unhomogeneous distribution of the activity, and θ_{a2} - the vertical. In this very case, θ_{a1} is evaluated 45 %, and θ_{a2} - 15%.

2.3. The activity distribution assessment in the body

As it was already pointed, the relation of the efficiency to the activity distribution in the body and to the measurement conditions, enables the qualitative analysis of the activity distribution

For that purpose, the F(E) factor is introduced, and:

$$F(E) = \left[q_s(E) / q_m(E)\right] \pm \delta F$$

where $q_s(E)$ is the full-energy peak net area counting rate (with energy E), when measuring a man without motion, the detector being fixed above the chest;

$q_m(E)$ is the full- energy peak net area counting rate (with energy E) , for a man's standard measurement with motion.

$$\delta F = \sqrt{\delta q_s^2 + \delta q_m^2}$$

When the geometric measurement conditions (distance to the detector and size of the measured object) are fixed, F(E) will depend on the activity distribution in the human body. In this case (r = 6 cm), if F(E) < 1,5, it is assumed that the activity evaluated A(E) is homogeneously distributed. If F(E) > 2,5, the activity is considered to be concentrated in the lungs.

2.4. The incorporated activity evaluation in real measurements

On purpose to test the above described methods, the following measurements were completed:

- measuring the background with a "zero" - phantom, imitating a standard man and containing 600 g KMnO S14 T (4500 Bq K-40);
- measuring the persons X and Y (workers from NPP-Kozloduy - Part 2). X was measured in standard conditions (with motion) and Y - in both kinds of measurement conditions.

The results from the measurement with the "zero"- phantom, are shown in table 4. The evaluations for the respective "critical" levels, detection levels and detection limits of the corresponding radionuclides, are also given.

Table 4.

param.	200-2000 keV	662 keV	1332 keV
q_b (E),imp/s	2.530	0.0130	0.00165
σ_b (E),imp/s	0.007	0.0006	0.00028
R(E), imp/s	0.080	0.0070	0.0032
D(E), imp/s	-	0.0160	0.0075
UDA(E), Bq	-	175	110

Notice: The counting rate in the 662 keV and 1332 keV gamma-lines is due to the Cs-137 and Co-60 pollution of the shield.

. The counting rate in the 200 - 2000 keV energy interval, registrated when measuring X with standard motion, was 2.17 imp/s, that is for X it is assumed that A(Cs-137) < 175 Bq and A(Co-60) < 110 Bq.

In table 5, the results from the measurement of Y, are shown. The counting rates in the respective energy intervals for the kinds of measurement conditions qs(E) and qm(E), the activity evaluated according to 2.2, the uncertainty confidence level and the F(E) factor evaluation, are also given.

Table 5

param.	200-2000 keV	662 keV	1173 keV	1332 keV
q_m (E), imp/s	9.25	0.108	0.0385	0.0344
σ_m (E), imp/s	0.06	0.008	0.0046	0.0058
A(E),Bq/body	-	1050	480	460
σ_A , %		±30%	±30%	±30%
q_s (E), imp/s	-	0.157	0.108	0.0924
F(E)		1.4 ±9%	2.8 ±14%	2.7 ±15%

That is, the radionuclides incorporated in Y's body, are: Cs-137 with 1050 Bq and Co-60 with 470 Bq activity. The total uncertainty of the results is determined mainly by the systematic uncertainty, and is evaluated to 30 % (for 68 % confidence level).

The F(E) evaluation shows that tha Cs-137 radionuclide activity is homogeneously distributed, while the Co-60 activity is concentrated in the lungs.

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BG9700059

SUMMARY ASSESSMENT OF THE RADIOBIOLOGICAL SIGNIFICANCE OF RADIONUCLIDES IN THE CONTAMINATION CONTENT PROCESSING GREAT NUMBER OF MEASUREMENTS

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Determination of the quality and quantity content of the radioactive contamination in NPP gives information about the origin of this contamination (presence of leakage in different systems, the characterization of technological processes at reactor refueling and repair operations, etc.), as well as - which is of primary importance - of the potential danger they would be for the staff and environment. The real danger is present when they would fall possibly in human body by air, through lungs and to a certain extent (practically zero) - through the digestive system. The Radiation Protection Codes (RPC-92)[1] accepted in our country in 1992 determine the maximum permissible concentrations of the content of every individual radionuclide in air and water for the categories:

- „A", persons exposed to occupational irradiation or involved in emergency operations;
- „B", separate persons or limited groups of population;
- „C", the country population as a whole.

In these Codes the permissible radioactive contamination of different surfaces concerning the total alpha- or beta-particles are given for category „A"[1].

As a practice in NPP are investigated radioactive contaminations precipitated on different surfaces in the technological rooms and their content in air of the ventilating system. The radionuclide identification is comparatively easy implemented when the radionuclides are gamma-emitters by means of a high-resolution gamma-spectrometer. The identification of alpha- and especially of beta-emitters is more difficult and could be completed only in separate cases.

Our experience in investigation of the radioactive contamination with radionuclides in the technological rooms of NPP „Kozloduy" showed that obtaining quantitative results according to the RPC-92 requirements is a very difficult task. To solve the problem it is necessary to control continuously the content of various radionuclides in air at operation, refueling and repairing operations and in very many points of every room. This is simply impossible. On the other hand the identification of the radionuclides precipitated on various surfaces is comparatively easy implemented and this contamination is due in many cases to the precipitation of aerosols and hot particles from the air. Even when the surface contamination is a result of a radioactive solution leakage it also could be seen as a problem of content of radionuclides in air because after drying they could disperse.

The presence of a considerable number of results concerning surface contamination which could not be used directly for the assessment required by RPC-92 gives a hint that they could be rationalized if an unit for their radiobiological significance is introduced. As such an „unit" we choose the ^{60}Co activity existing in practice always. The relative radiobiological significance of i-radionuclide in relation to that of ^{60}Co is introduced as follows:

$$\eta_i = \frac{[\mathrm{MAPC}_{air}]}{[\mathrm{MAPC}_{air}]_{^{60}Co}} \qquad (1)$$

[1]Civil Defence Administration, Sofia
[2]Sofia University „St. Kliment Ohridski"

where $MAPC_{air}$ is the mean-annual permissible concentration in air for every one of them according to RPC-92. This value is put together with the experimentally measured one for certain sample:

$$\eta_{exp,i} = \frac{A_i}{A_{^{60}Co}} \qquad (2)$$

where A_i is the experimentally determined value of i-radionuclide activity and $A_{^{60}Co}$ - for ^{60}Co in the same sample (in Bq). It turned out that η_i is equal or very close in value for groups of radionuclides which could be seen in the accompanying table. That's why in determining $\eta_{exp,i}$ it is necessary to get the sum of the activities of that group whose terms we indicate with j:

$$\eta_{exp,i} = \frac{\sum_j A_{j(i)}}{A_{^{60}Co}} \qquad (3)$$

The relation:

$$R_i = \frac{\eta_{exp,i}}{\eta_i} \qquad (4)$$

shows how many times the activity of i-radionuclide (respectively the radionuclide group according to (3)) is more significant than that of ^{60}Co from a radiobiological point of view.

When there are a lot of measurements of samples from one and the same room (or when averaging in other symptom) the average value of $\eta_{exp,i}$ is:

$$\bar{\eta}_{exp,i} = \sum_n \left(\eta_{exp,i,n} \cdot \frac{A_{^{60}Co,n}}{\sum_n A_{^{60}Co,n}} \right) = \frac{\sum_n A_{i,n}}{\sum_n A_{^{60}Co,n}} \qquad (5)$$

where $A_{i,n}$ refers to the sample under No.n for which the ^{60}Co activity is $A_{^{60}Co,n}$ and $\frac{A_{^{60}Co,n}}{\sum_n A_{^{60}Co,n}}$ is the „relative weight" of the n-sample.

This approach of radiobiological significance evaluation of certain group (in the meaning of (3)) is reasonable only if the ^{60}Co activities prevail considerably over the remaining which is observed to the main part of the contamination in NPP „Kozloduy". In contamination of other kind the radionuclide whose activity prevails over the activity of the remaining radionuclides could be used for comparing in the same way.

It is convenient the results of the third table column to be plotted on nomograms, examples of which are given in Fig.1 to 5. The logarithmic value of η_i is plotted on the nomogram abscissa and that of R_i on the ordinate. The abscissa zero point correspond to the position of ^{60}Co. The benefit of the method proposed here for estimation of radiobiological significance of various radionuclide groups could be best understood by the conclusions made on the basis of the nomograms concerning different objects in NPP „Kozloduy", namely:

1. In Fig.1 is seen that after the repair operation in the technological rooms of Unit I the alpha-emitter contamination is increased considerably. The contamination of other radionuclide groups compared with that of ^{60}Co could be neglected (having in mind the short half-decay times of the most of them). This conclusion is valid for the remaining nomograms.

2. It is surprising that in Unit IV the relative alpha-nuclide contamination is considerably much more than that of Unit I which is seen in Fig.2. As is well-known a large content of ^{110m}Ag is available here.

3. Fig.3 shows that the alpha-radionuclide content of Unit V (WWER-1000) is negligible compared with that of WWER-440 units.

4. The results in Fig.4 are very indicative. The radionuclide content in the air filter of the reactor room of Units I and II is entirely comparable with their content in air concerning ^{60}Co. Air samples are taken outside the premises, from the V floor of a building situated approximately 50 m away of the Unit (SBK-2). (The content absolute value of these radionuclides at NPP on-site is hundreds of thousands to billions times lower than the permissible concentrations for the population).

5. A special case is presented in Fig.5 concerning a single sample - radionuclide contamination of working dress. As the radiobiological significance of some radionuclides exceed considerably that of ^{60}Co they are pointed purposely in the groups of which they belong.

We consider that using this method one could make essential conclusions both to the contamination origin and the countermeasures about the radionuclides representing a risk for the staff. It is worth to note that the value of η_i is one and the same for persons of categories „A", „B" and „C" pointed in RPC-92.

Acknowledgment

We take advantage of the opportunity to express our gratitude to the NPP „Kozloduy" Administration for the moral and material support.

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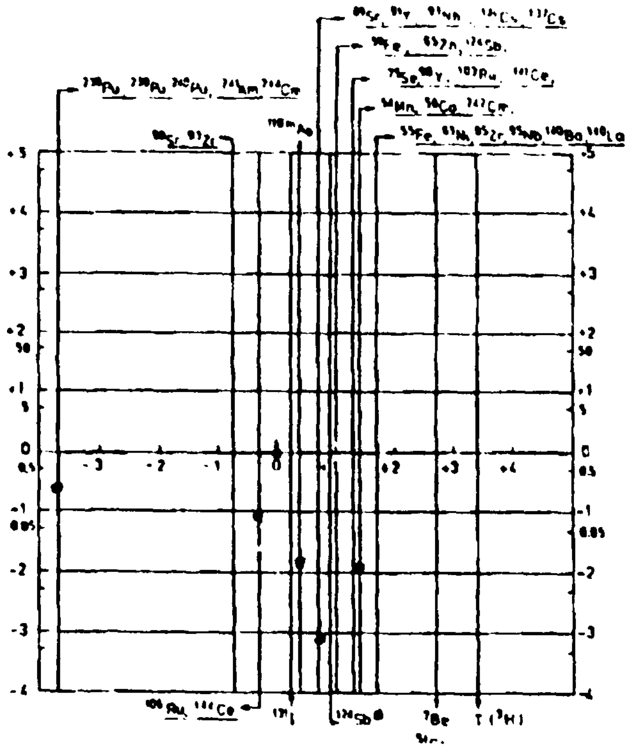
Table certifying the radioactive contamination in relation to ^{60}Co content

η_i	$\eta_{exp,i}$	R_i	Radionuclides
2.10^{-4}			^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{244}Cm
0.13			^{90}Sr, ^{93}Zr
0.5			^{106}Ru, ^{144}Ce
1			^{60}Co
1.74			^{131}I
2.5			^{110m}Ag
5			^{89}Sr, ^{91}Y, ^{93}Nb, ^{134}Cs, ^{137}Cs
10			^{59}Fe, ^{65}Zn, ^{124}Sb
20			^{75}Se, ^{90}Y, ^{103}Ru, ^{141}Ce
25			^{54}Mn, ^{58}Co, ^{242}Cm
50			^{55}Fe, ^{63}Ni, ^{95}Zr, ^{95}Nb, ^{140}Ba, ^{140}La
500			^{7}Be, ^{51}Cr
2500			$T(^{3}H)$



Unit I, August 1992, surface contamination

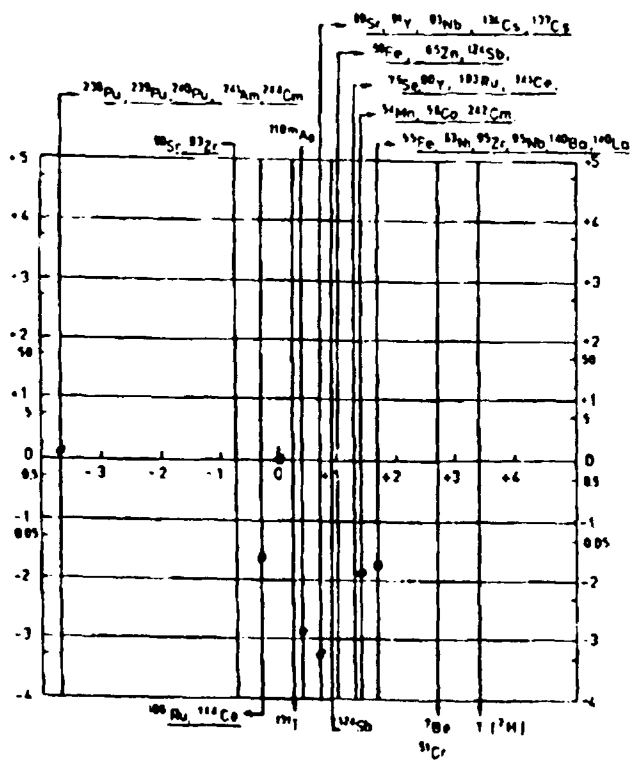


Fig. 1 Unit I, March 1993, surface contamination at repair operations

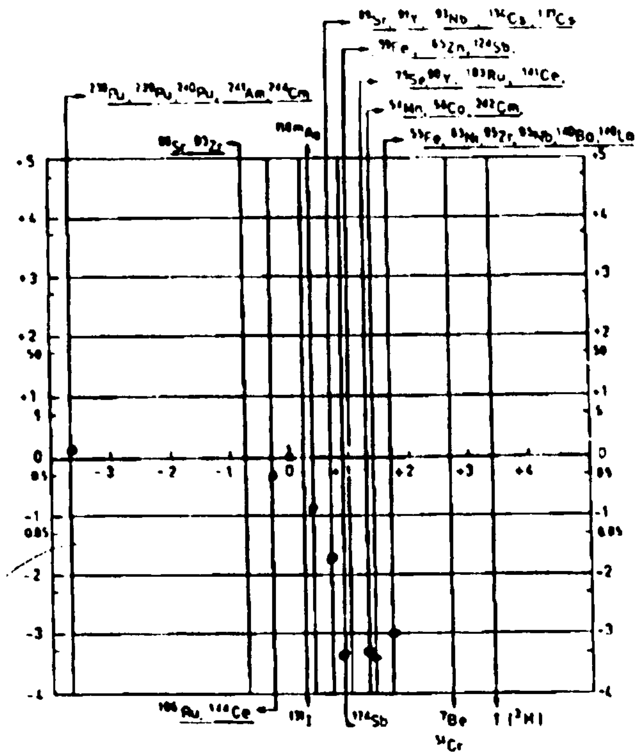


Fig. 2 Unit IV, February 1993, surface contamination

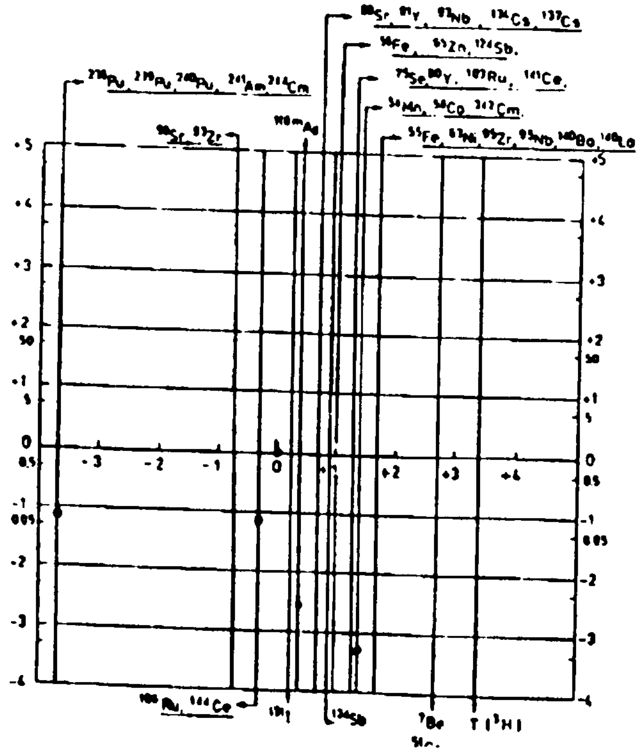


Fig. 3 Unit V, August 1992, surface contamination

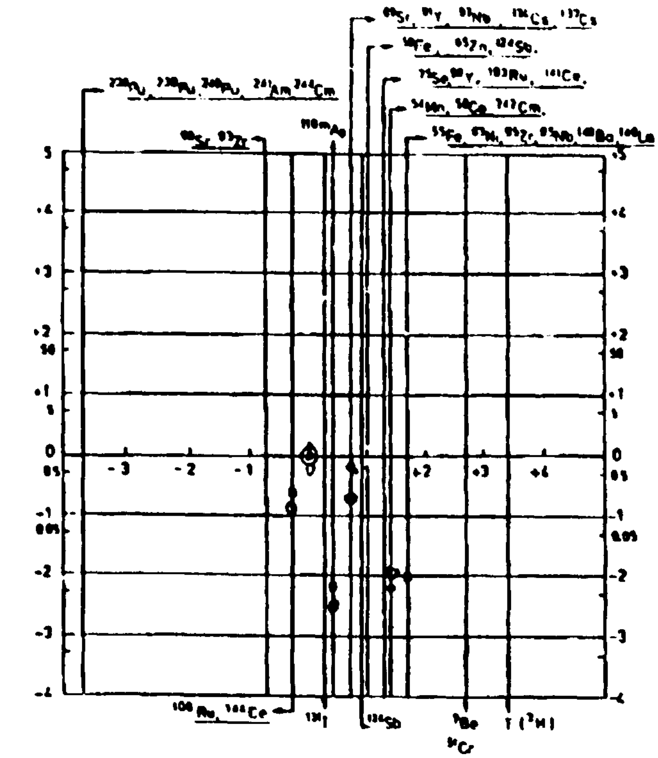


Fig.4 • Air filters of the reactor room of Units I and II and SBK-2, June 1992
o Air samples from EP-1 sampled at the same time
The alpha-radionuclide content is not evaluated

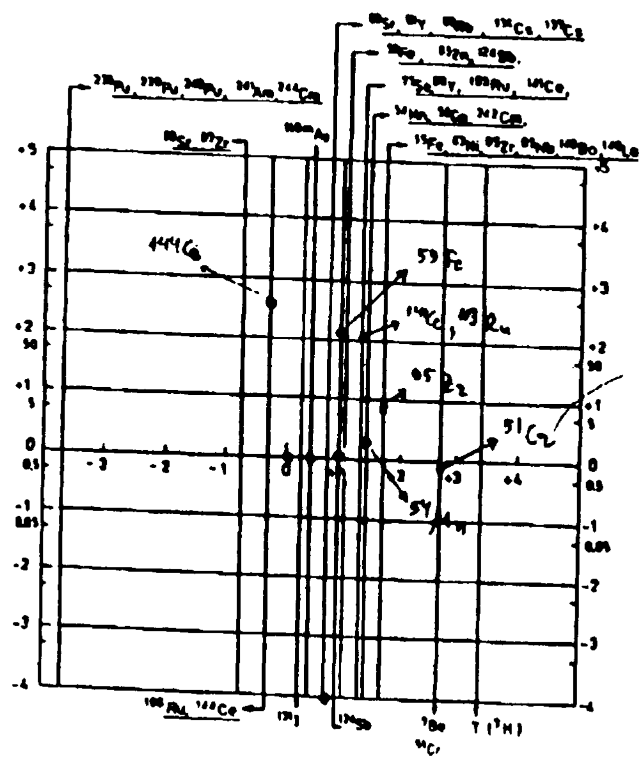


Fig. 5 Working dress used at refueling operation of Unit IV
The alpha-radionuclide content is not evaluated

BG9700060

ЗАБОЛЕВАЕМОСТЬ ЗЛОКАЧЕСТВЕННЫМИ НОВООБРАЗОВАНИЯМИ СРЕДИ НАСЕЛЕНИЯ В РАЙОНЕ АЭС "КОЗЛОДУЙ"

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Один из основных эффектов действия низких доз йонизирующих излучений являются злокачественные новообразувания. Публикация 60 Международной комисии радиологичной защиты определяет самый большой коэффициент риска для возникновения злокачественных новообразуваний, при воздействие йонизирующих излучений, для легких, молочной железы, щитовидной железы и лимфной и кровтворной ткани. Всемирная организация здравоохранения и Международная агенция атомной энергии предлагают при изучение поздных эффектов влияния низких доз, кроме других известных показателей, самое большое внимание отделять злокачественным новообразуваниям щитовидной железы и кроветворной ткани, на ранних стадиях, и злокачественным новообразувания легких и молочной железе на поздних стадиях после облучения. В связи с этом, была исследована заболеваемость злокачественными новообразуваниями легких, молочной железы, щитовидной железы, лимфной и кроветворной ткани в 30 км. зоне вокруг АЭС "Козлодуй". Полученные результаты были сравнены с районами вокруг Белене и Ямбол.

Прилегающий район вокруг Козлодуя хорошо исследован в радиационном отношение. С 1974 года в АЭС "Козлодуй" функционируют реакторы ВВЕР. Район города Белене имеет схожее географическое расположение, численность и структура населения. В районе Белене началось сооружение второй АЭС в Болгарии, но пока оно приостановлено. Естественный радиационный гамма фон в районе Белене, находится в границах средних показателей для Болгарии. В районе города Ямбол, стойности естественного радиационного гамма фона выше чем в Болгарии. Ямбол имеет схожая численность и структура населения как район Козлодуй. Полученные результаты были сравнены с средними показателями для Болгарии, чтобы определить наличие или отсутствие каких-либо тенденций в развитие заболеваемости злокачественными новообразуваниями в данных районах.

Для изучения заболеваемости злокачественными новообразованиями источником информации была Учетная форма N° 58 - где регистрируется каждое заболевание злокачественным новообразованием, с указанием диагноза, пола, возраста заболевшего. Необходимые данные для численности населения были получены из Национального статистического института.

Мы провели ретроспективное исследование заболеваемости злокачественными новообразованиями легких, молочной железы, щитовидной железы, лимфной и кроветворной ткани за период 1985-1990 г. в указанных выше районах. Мы вычислили фактическую и стандартизированную заболеваемость по возрастным группам и полу. Цель стандартизации - устранить разницу в возростных структурах населенных пунктов и получить более четкое представление о совместном влиянии остальных рисковых факторов на уровень заболеваемости. Мы использовали мировой стандарт для стандартного населения, который был предложен Международной агенции для борьбы с раком. Была вычислена стандартная ошибка и доверительный интервал с гаранционной вероятностью 68% и 95%.

Стандартизированная заболеваемость раком легких на период 1985-1990 для всего населения самая высокая в районе Козлодуя - 21,19 ± 1,33 (фиг.1). Разница с районом Белене (17,45 ± 1,27) статистически достоверна, так как с районом Ямбола

(18,73 ± 1,14) она не достоверна. При этом все три района (вкл. Козлодуй) имеют более низкую заболеваемость чем в среднем для Болгарии - 21,82 ± 0,42. Для мужчин самую высокую заболеваемость определяется в районе Козлодуя - 38,87 ± 2,62, которая выше чем в районе Белене - 32,72 ± 2,42 и выше чем в районе Ямбола - 35,27 ± 2,25. Опять же все три районы имеют заболеваемость ниже чем в стране, которая 40,51 ± 0,82. Для женщин тоже район Козлодуя имеет самую высокую заболеваемость - 5,5 ± 0,9. Эта стойность выше чем в стране, которая 4,9 ± 0,27, но разница между ними недостоверна. При изучение заболеваемости по районам, мы не определяем каких-либо тенденций в ее динамике.

Рак молочной железы показан на фиг. 2. Стандартизированный показател на период 1985-1990 г. самый высокий в районе Козлодуя - 36,23 ± 1,99. Он достоверно выше чем в районе Белене (30,27 ± 2,57) и районе Ямбола (24,89 ± 1,98), но ниже чем в стране (37,37 ± 0,81). По годам определенную тенденцию в развитие заболеваемости не определяется.

Стандартизированный показатель заболеваемости злокачественными новообразованиями лимфной и кроветворной ткани самый высокий в районе Козлодуя - 7,54 ± 1,13 и в районе Белене 7,36 ± 0,95, но при этом разница между ними недостоверна. Ямбол имеет самую низкую заболеваемость - 4,42 ± 0,59 и разница с районами Козлодуя и Белене и со средними в стране статистически достоверна. Анализ по годам показывает, что в районе Козлодуя в период 1985-1990 г. сначала определяется уменьшение показателя - 9,7 - 8,4 - 7,0 - 5,9, который в конце периода повышается. В районах Белене и Ямбола не определяется тенденцию в динамике заболеваемости.

Заболеваемость раком щитовидной железы в Козлодуйском районе - 1,28 ± 0,41, в Беленском 1,12 ± 0,39. Оба района имеют более низкие показатели, чем в среднем в стране, который 1,36 ± 0,12. В районе Ямбола за весь период исследования не имеется ни одного случая заболевания раком щитовидной железы среди мужчин. Поэтому не определяется заболеваемость для всего населения.

(Все показатели заболеваемости злокачественными новообразованиями определены на 100 000).

Выводы:

1. Заболеваемость злокачественными новообразованиями легких в районе Козлодуя выше чем в районе Белене. В целом, все районы (Козлодуй, Белене, Ямбол) имеют достоверно более низкую заболеваемость чем в среднем в стране.

2. Заболеваемость раком молочной железы во всех районах достоверно ниже, чем в среднем в стране. По годам не определяется определенную тенденцию.

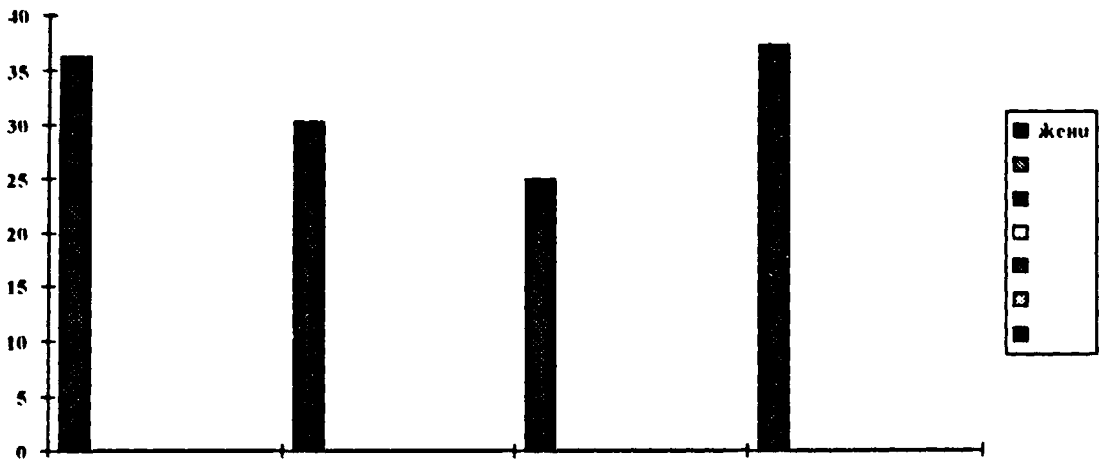
3. Заболеваемость злокачественными новообразованиями лимфной и кроветворной ткани в районе Козлодуя выше, чем в стране. Но при этом разница статистически недостоверна.

4. Заболеваемость раком щитовидной железы в районах более низкая, чем в стране. Число случаев очень малое, что не позволяет сделать более подробный анализ по годам.

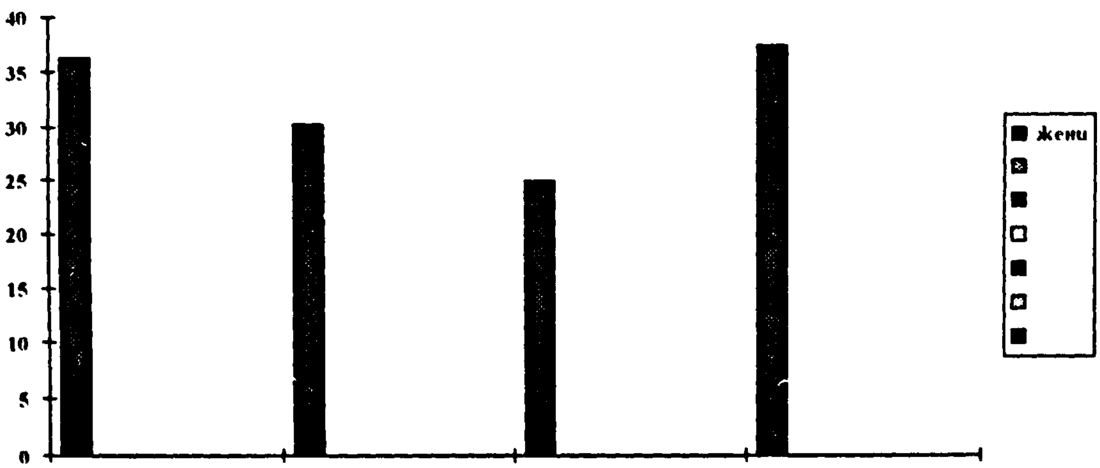
5. Наблюдаемые разницы в уровень заболеваемости в районах находятся в границах естественных вариаций исследуемых показателей.

фиг. 1 Стандартизированные показатели заболеваемости злокачественными новообра-зованиями легких (на 100000 чел.) за период 1985 - 1990 гг. по районам

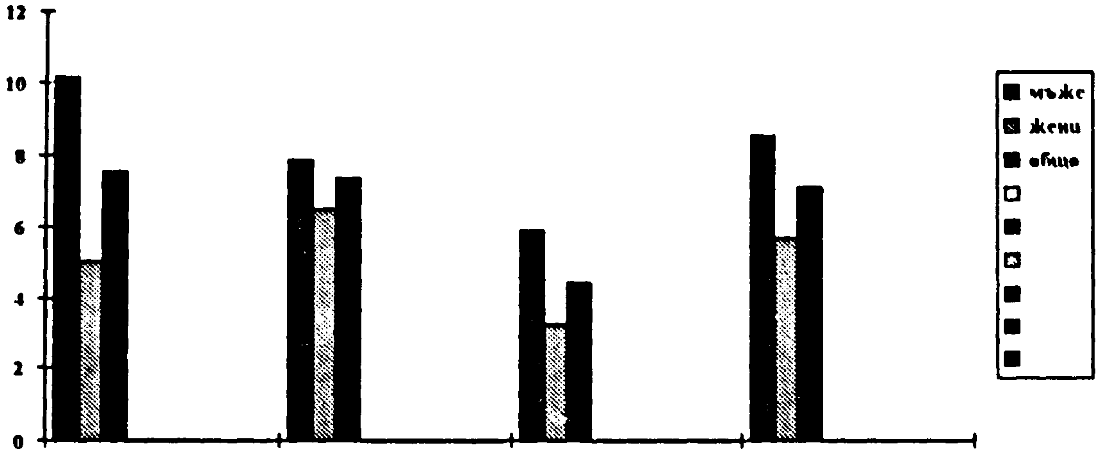
Козлодуй Белене Ямбол България



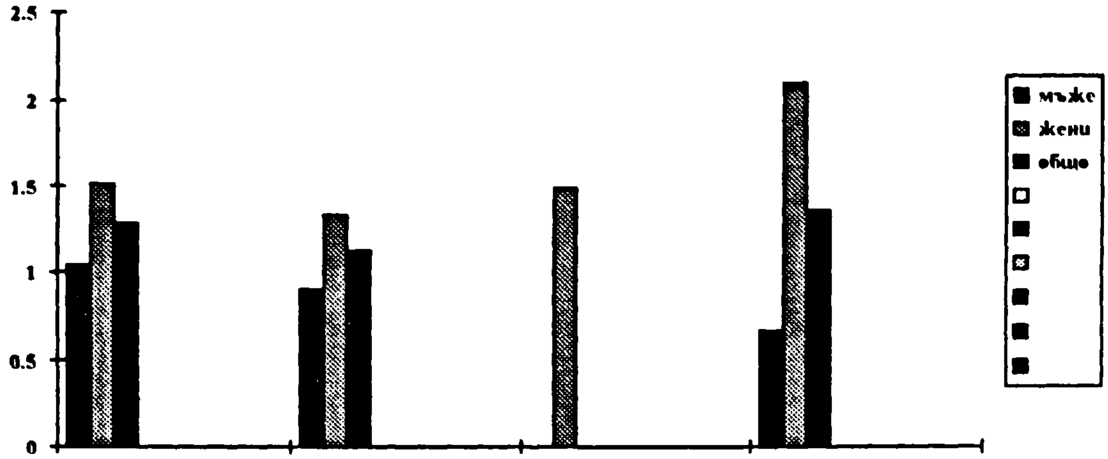
фиг. 2 Стандартизированные показатели заболеваемости злокачественными новообра-зованиями молочной железы (на 100000 чел.) за период 1985 - 1990 гг. по районам



фиг. 3 Стандартизированные показатели заболеваемости злокачественными новообра-зованиями лимфатической и гемопоетичной ткани (на 100000 чел.) за период 1985 - 1990 гг. по районам



фиг. 4 Стандартизированные показатели заболеваемости злокачественными ново-образованиями щитовидной железы (на 100000 чел.) за период 1985 -1990 гг. по районам



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Содержание 137Cs и 90Sr в объектах окружаlющей среды в районе АЭС "Козлодуй"

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Анотация: Проанализирован радиационный статус наиболее биологически значимых радионуклидов техногенного происхождения 90Sr и 137Cs, которые были определены в пробах почвы и растительности за период 1990-1993 гг. в наблlюдаемой и контролируемой зонах вокруг функционирующей АЭС "Козлодуй". Из полученых результатов радиоэкологического исследования вытекает, что средногодовые стойности следуlют ходу динамики радионуклидов по годам, а неперерывная эксплуатация АЭС "Козлодуй" не влияет на радиационный статус окружаlющей среды.

Радиоэкологические исследования содержания техногенных радионуклидов в объектах окружаlющей среды составляlют один из способов контроля радиационной безопасности населения путем изучение поведения глобальных атмосферных выпадений и влияния функционирующей АЭС "Козлодуй" на загрязнение внешной среды. Эти исследования проводятся ежегодно с цельlю оценки величины радиоактивного загрязнения вследсвии эксплуатации атомной электростанции как часть повсеместного и неперерывного контроля исскуственой радиоактивности.

Цельlю настоящего сообщения явилось получение польной информации содержания наиболее биологически опасных радионуклидов 137Cs и 90Sr в объектах окружаlющей среды: почвы и растительных образцах в районе наблlюдаемой и контролируемой зонах около АЭС "Козлодуй" за период 1990-1993 г.г.

Материалы и методы

Секция "Радиоэкология" к НЦРРЗ-София проводит контроль радиационного статуса от начала пуска в эксплуатациlю АЭС "Козлодуй" (1968-1974). Данные собраны в этот период послужили для базы при оценки влияния функциониращей АЭС [2,3,4].

В настоящей разработке разсматривается поведение техногенных радионуклидов 137Cs и 90Sr, которые имеlют самый длительный период полураспада.

Взяты и выбраны образцы почвы и травы - элементы биогеоценозной цепи, так как почва - самый чувсвительный индикатор радиоактивных выбросов в атмосферу, а трава - главное звено проникновения радионуклидов в молоко и мясо. Каждый сезон берутся пробы для исследования из стационарных пунктов в зоне факела АЭС, которая охватывает 12 км контролируемуlю зону с пунктами Оряхово, Бутан, Гложене, Козлодуй, Крушовица и 30-км наблlюдаемуlю зону с пунктами Червен бряг, Пелово, Плевен и Гиген. Контрольным пунктом является город Лом, который находится в 40 км от Козлодуя в направление северо-северозапад [1]. Схема района и пунктов пробоотбора показана на фиг. 21.

Взятые пробы подвергаlются предварительной обработке, после чего проводится радиохимический анализ с последуlющим радиометрированием 137Cs и 90Sr.

Результаты и обсуждение

Среднегодовые концентрации 90Sr и 137Cs в почве и траве в пунктах контролируемой зоны и контрольного пункта - город Лом представлены на фиг. 1-12. Как общая закономерность наблlюдается значительно по-ниская концентрация

137Cs в траве от 0 до 5 Bq/kg, в сравнении с почвой, где показатели варирует от 0 до 140 Bq/kg. Исключением являются показанные на фиг.5 и фиг.9 содержания 137Cs в траве - 12.6 Bq/kg и 19 Bq/ kg соответственно.

Исследования показывают, что содержание 137Cs в почве значительно превышает содержание 90Sr, что доказывает постоянного связывания и накопления цезия. Это наблюдается во всех исследованных почвах при максимальной стойности 90Sr - 3.78 Bq/kg (Крушовица, 1991г.) , а 137Cs -144 Bq/kg (Оряхово, 1992 г.).

При сравнение концентрации двух радионуклидов в траве не установлени значительние различия между стоиностями (0 - 6 Bq/kg) в отличие от почвой.

Интересен тот факт, что в некоторых пунктах (фиг. 2, 6, 10, 12) годовое содержание 90Sr в пробах травы значительно высше чем в почве.

Среднегодовые стойности концентрации 137Cs и 90Sr во всех пунктах в контролируемой зоне сравнены с этими в наблюдаемой зоне и в контрольном пункте г.Лом (фиг. 13-16), при этом фиг. 13-14 относятся для почвах, а фиг.15-16 - для пробах травы. Оказывается что концентрации двух радионуклидов в прочве выше в наблюдаемой зоне чем в контролируемой. В контрольном пункте как ожидалось стоиности остаются более нискиe. Это тенденция не наблюдается при травах. Начиная с 1991г. для 137Cs (фиг. 15) и с 1992 для 90Sr (фиг. 16) регистрированые значения более высокие в контрольном пункте. С целю установить где самое большое накопление 137Cs и 90Sr по годам, мы графично сравнили их показатели в травах и почвах во всех пунктах исследованного периода (фиг. 17, 18, 19, 20). Самые высокие концентрации радиоцезия в почве мы нашли в Оряхово, Гиген и Пелово (фиг. 17), а для 90Sr в Оряхово, Крушовица и Пелово. Стойности двух радионуклидов в травах были относительно одинаковые для всех пунктах, с небольшими исключениями для 137Cs в отделных годах (Бутан -1990, Лом-1993, Гиген 1990). Для 90Sr наблюдается слабое повышение концентрации в 1992 и 1993 годах и в двух объектах.

Выводы

1. Сделаные исследования радиацинного контроля внешней среды в районе АЭС "Козлодуй" на периоде 1990 - 93, показывают что нормальное функционирования АЭС не оказало заметного влияния на радиоактивность наблюдаемых объектов биогеоценозной цепи.

2. Целостный анализ полученых данных содержания 90Sr и 137Cs в рассматриваемых объектах дают нам основание считат, что выбросы в атмосферу при нормальной эксплуатации АЭС этих долгоживущих радионуклидов бывают в количествах каторые на фоне глобальних радиоактивних отложении в контролируемом районе не могут быт отдиференцированными с помощью наличной аппаратуры и методы контроля.

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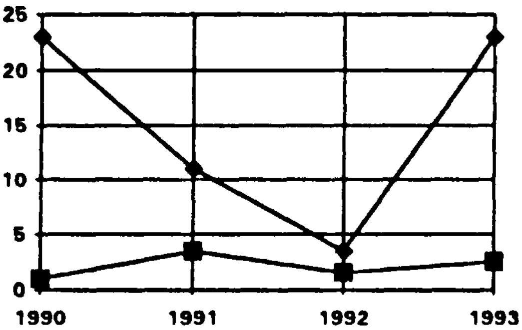
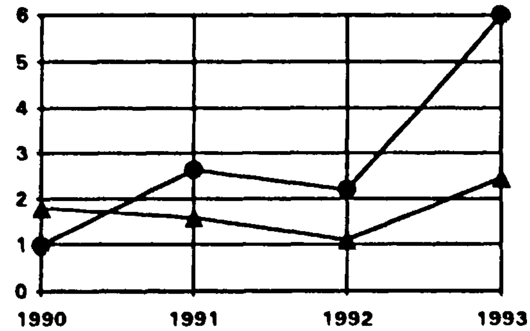
Козлодуй

Cs 137
фиг. 1

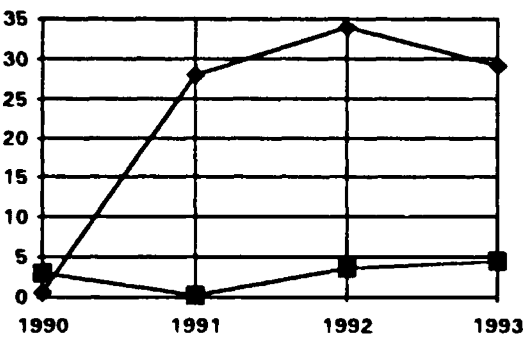


Sr - 90
фиг. 2

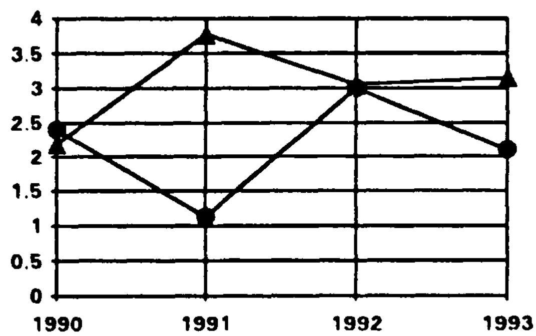


Крушовица

Cs - 137
фиг. 3

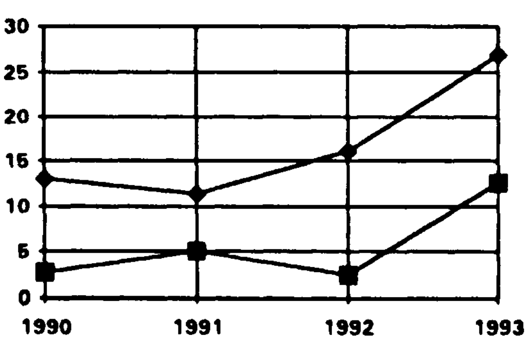


Sr - 90
фиг. 4

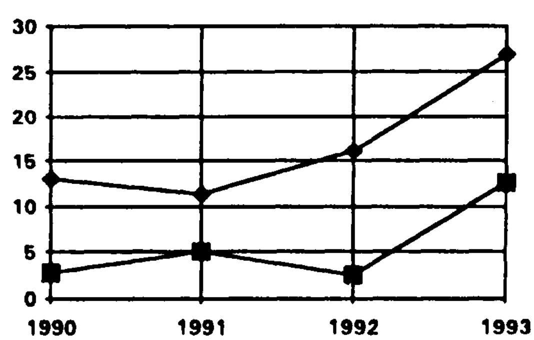


Лом
контрольный пункт

Cs - 137
фиг. 5



Sr - 90
фиг. 6



Легенда:

◆ - концентрация Cs - 137+134 в почве

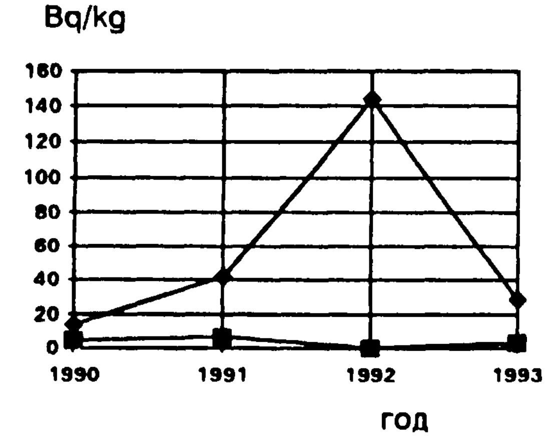
Δ - концентрация Sr - 90 в почве

■ - концентрация Cs - 137+134 в треве

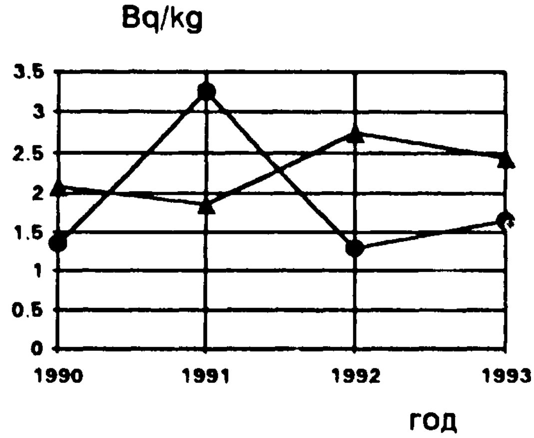
● - концентрация Sr - 90 в треве

Оряхово

Cs -137
фиг. 7

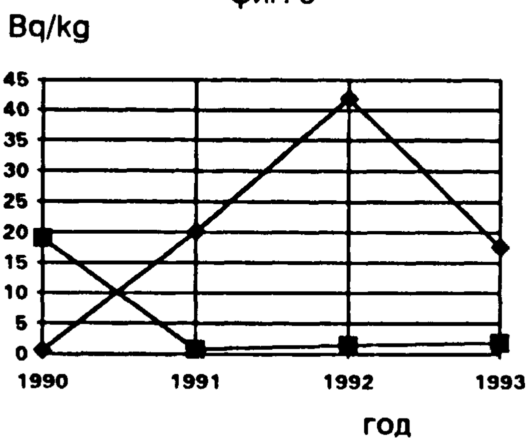


Sr - 90
фиг. 8

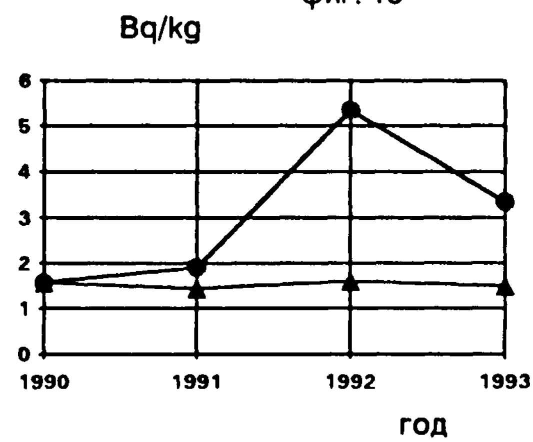


Бутан

Cs 137
фиг. 9

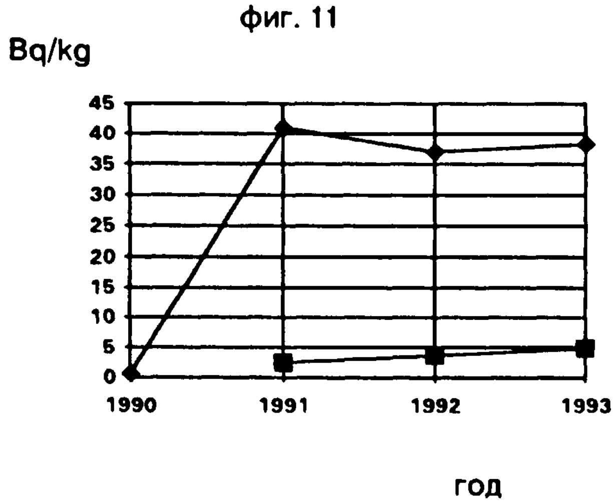


Sr - 90
фиг. 10

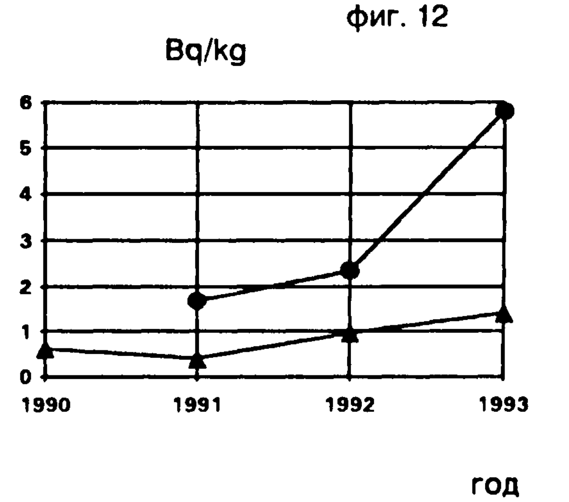


Гложене

Cs 137
фиг. 11



Sr - 90
фиг. 12



Легенда:

◆ - концентрация Cs - 137+134 в почве

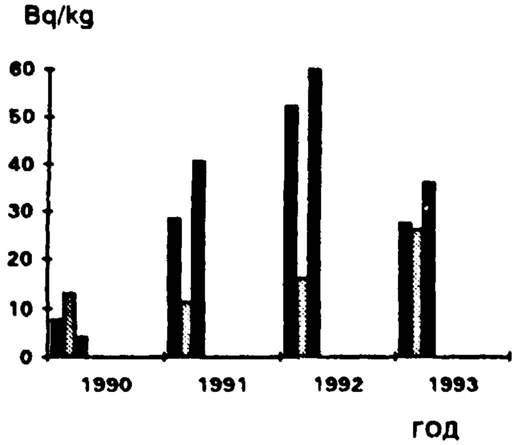
■ - концентрация Cs - 137+134 в треве

Δ - концентрация Sr - 90 в почве

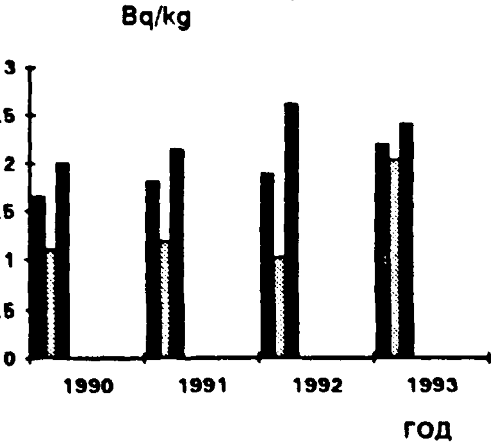
● - концентрация Sr - 90 в треве

ПОЧВА

Cs -137
фиг. 13

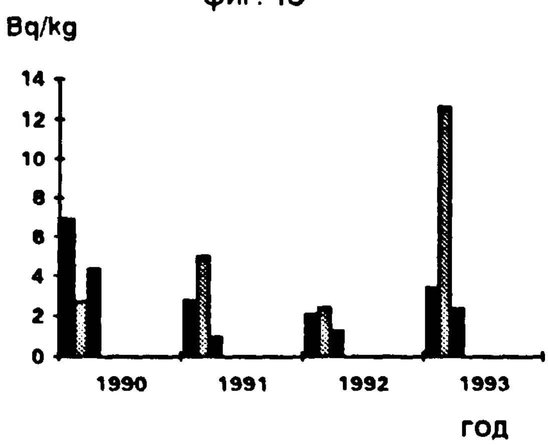


Sr - 90
фиг. 14

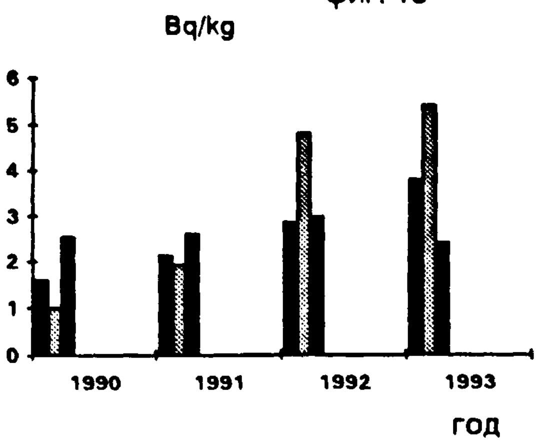


ТРАВА

Cs -137
фиг. 15



Sr - 90
фиг. 16



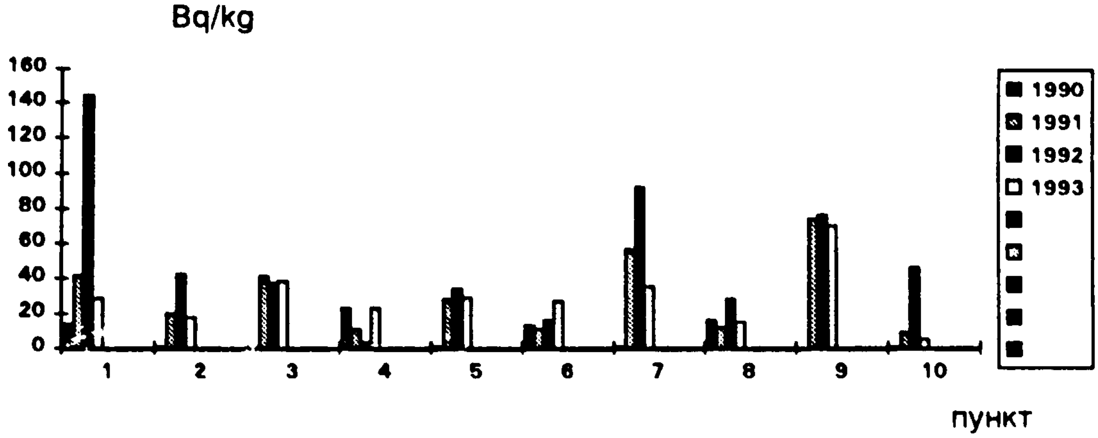
ЛЕГЕНДА

контролируемая зона

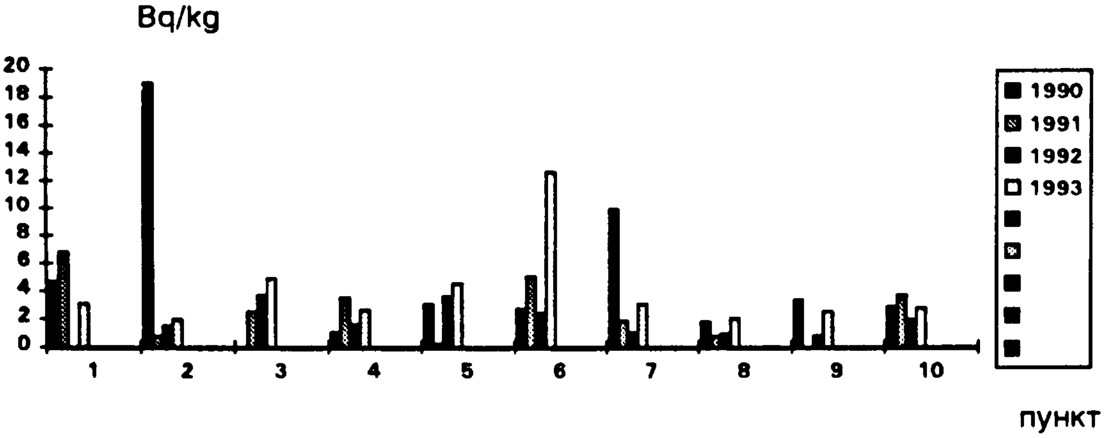
контрольный пункт

наблюдаемая зона

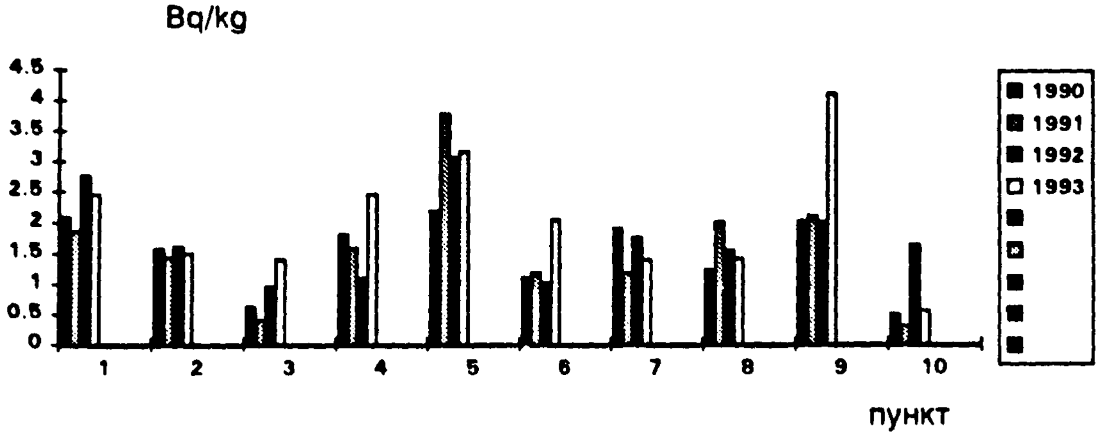
ПОЧВА
Cs - 137
1990 - 1993 г
фиг. 17



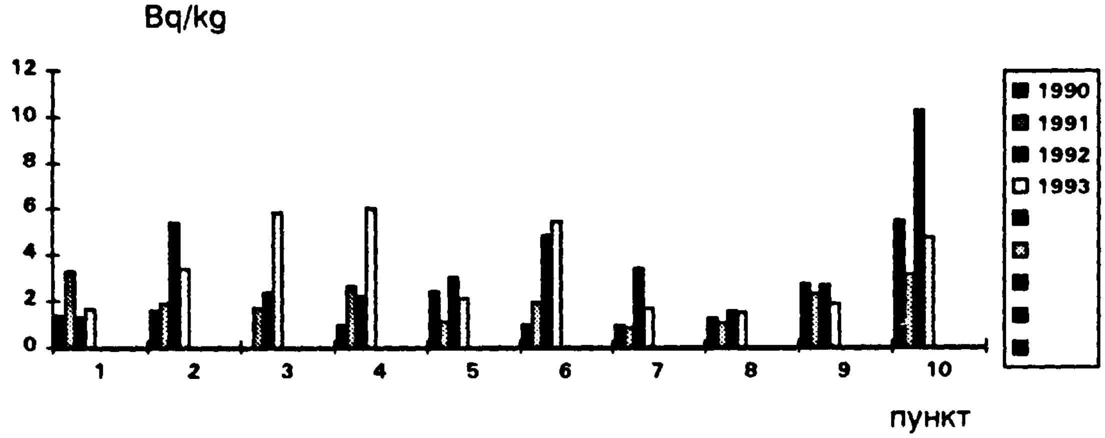
ТРАВА
Cs - 137
1990 - 1993 г
фиг. 18



ПОЧВА
Sr - 90
1990 - 1993 г
фиг. 19



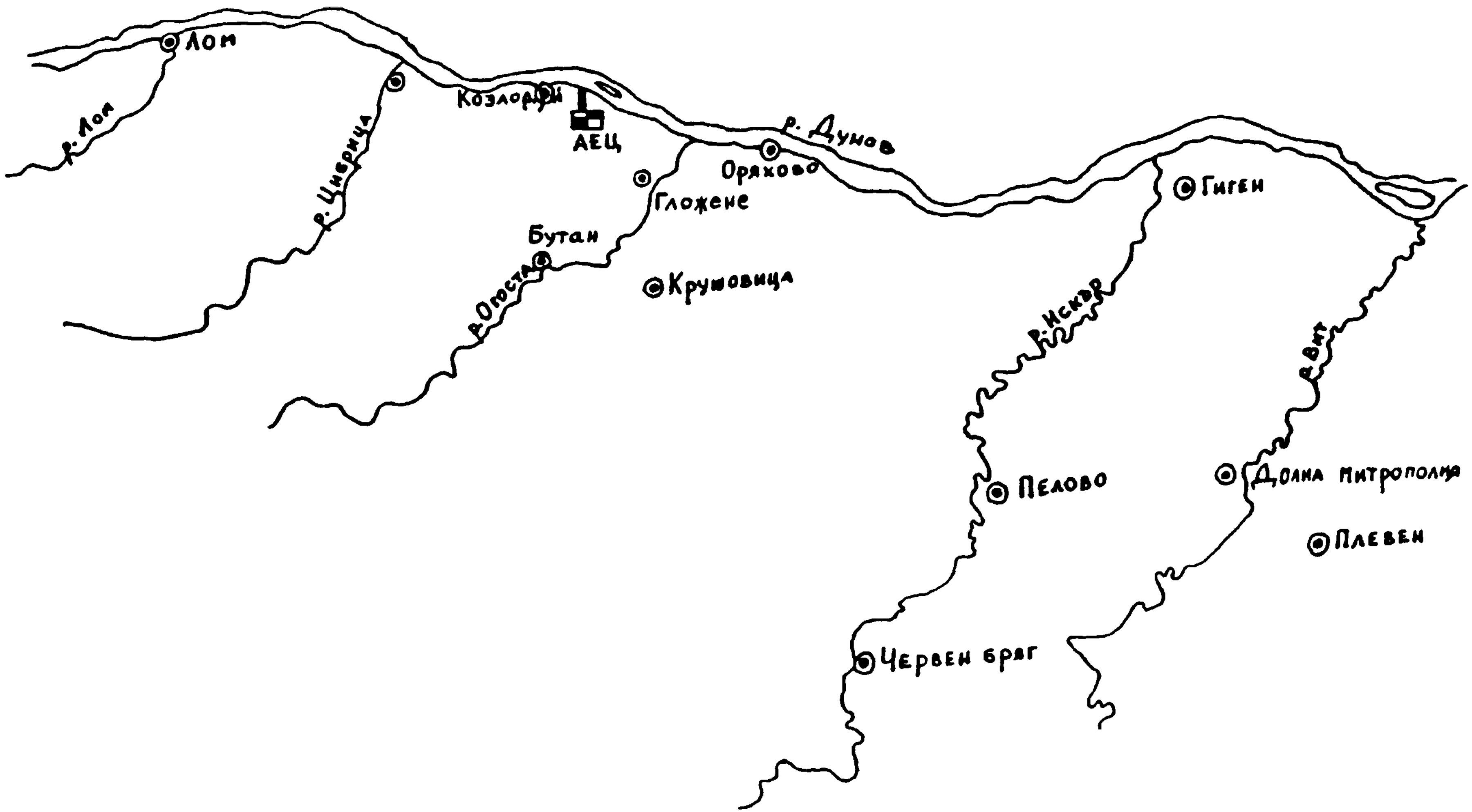
ТРАВА
Sr - 90
1990 - 1993 г
фиг. 20



ЛЕГЕНДА:

1. Оряхово
2. Бутан
3. Гложене
4. Козлодуй
5. Крушовица
6. Лом - контрольный пункт
7. Гиген
8. Плевен
9. Пелово
10. Ново село - контрольный пункт

фиг. 21 Контрольные пункты пробоотбора ●



CONCLUDING SPEECH

by the President of the Organization Committee

Prof. Dr. Tsvetan Bonchev

DEAR COLLEAGUES,

We took part in a non-trivial conference. I do not know if conferences like this are carried out in other nuclear plants. The countries developing nuclear energetics have large institutes and experimental bases which organize such meetings of experts including regular domestic and international conferences about different problems of the nuclear energetics. As it is known we also have specialized institutes and organizations in this field: the Nuclear Plants Department of ENERGOPROECT, some departments of the Institute of Nuclear Researches and Nuclear Energetics of the Bulgarian Academy of Science, the National Center of Radiobiology and Radiation Protection which carry out domestic and international seminars, conferences and meetings on the matters of the nuclear safety, problems with the metal, radioactive wastes, nuclear fuel, radioactivity in the environment etc. Some of these meetings have become traditional.

At the current conference in Kozloduy all subjects mentioned above have been presented and this was worthwhile. Experts from fields which as different they might seem but having re-covering problems and interests gathered for the first time and the benefit from that was doubtless. It is not less important that the conference was carried out in the town where the experts from the plant lived and were able to participate in the discussions of the complex problems of our nuclear energetics and to get in close contact with the scientific progress representatives from Bulgaria and other countries. There may hardly be doubt that the bilateral advantage from these contacts will be great. All this suggests that if the reason the current conference to be carried out was the celebration of the 20th anniversary of our Nuclear Power Plant it was worth to think whether a conference like this here in Kozloduy should not become a traditional one.

The present conference gave another result unforeseen by the organizers. The representatives of our scientific society including experts of the plant displayed the level of their researches and developments to their colleagues from other countries (Russia, USA, France, Japan) and we hope this level would be judged on its merits. This would affect the spirits and confidence for the future of the people involved in the nuclear energetics.

Now I take the liberty to pause on a ticklish point. Due to reasons which are well known and I am not going to set forth, the NPP Kozloduy became „world-famous" as nearly the most dangerous nuclear power plant and we know that this has nothing to do with the truth. The truth however is that the plant has its own serious problems which are better known by the people working in it than by anyone else, but unfortunately the solution of these problems is difficult because of the serious economic condition of Bulgaria as a whole and the NPP Kozloduy particularly. In relation with this we are sincerely thankful to the international organizations and separate advanced countries which render us assistance including material aid. However we cannot and we must not to rely only on help from outside, this leads to nothing good. One who counts only to outside assistance looses his independence and restrains his development. The latter does not suit to the capability of our science and technology. I think that it is better for us to aim to a much closer interaction and wider in range cooperation with the experts in the field of nuclear energetics from other countries.

Do we have real abilities to solve the problems of our nuclear energetics on our own which does not mean that we should not seek a cooperation with experts and organizations from other countries? In my view there are such directions and the evidence is presented at the current conference. I am speaking of:

- Probability assessments of the safety of the WWER reactors (RISKENGINEERING and ENERGOPROECT);
- Prognosis of the effect of an accident with a system of a particular type (ENERGOPROECT);
- Refueling optimization for effective utilization of the nuclear fuel (NPP Kozloduy, INRNE);
- Research of radioactive contamination in the technological rooms (NPP Kozloduy, SU);
- Radiological control of the environment (NPP Kozloduy, NCRRP, SU and other);

Irradiation of the personnel and the population, health consequences (NCRRP and NPP Kozloduy).

Of course there are many other more pressing tasks in areas where we have experts at our disposal also but in which we need to master the foreign experience, namely:

- Quality control of the metal of the reactor, steamgenerators and other (INRNE, NPP Kozloduy, IM);
- Problems with the radioactive wastes processing and storage (NPP Kozloduy - WESTINGHOUSE technology);
- Solution of the question with the spent nuclear fuel;
- Establishment of a national radioactive wast disposal facility (BAS).

Once again we would like to express our gratitude to the experts from different countries who took part in our conference. Let's hope that their presence here will be a serious prerequisite for establishment of a better international cooperation in the future.

I take the opportunity to express our common thankfulness to the management of the NPP Kozloduy and personally to its General Manager Dipl.eng. Kozma Kuzmanov not only for their enterprise the conference to be carried out but for the excellent conditions they created as well. We know that the plant experiences serious difficulties with the funds and accommodation but they created for us such a conditions of living that made us feel at home and this contributed to the successful conference implementation.

I wish you success in your future work. I hope we shall meet again.

26.10.94

Kozloduy