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CURRENT ABSTRACTS

Nuclear Reactors and Technology

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Nuclear Reactors and Technology

POWER REACTORS

Light-Water Moderated, Boiling Water Cooled

1
(BMU–1990-259)
Supplementary development and close-to-reality testing of a novel technique for complete ultrasonic inspection of the nozzle field of the spherical bottom of BWR pressure vessels. Brekow, G.; Wuestenberg, H.; Erhard, A.; Hein, H. (Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany); Bundesanstalt fuer Materialforschung und -pruefung, Berlin (Germany)). May 1990. 48p. Available from FIZ Karlsruhe.

The results of two measuring campaigns in a test stand using the phased array technique, and the subsequent on-site measurements in the spherical pressure vessel bottoms of the Brunsbuettel and Kruemmel reactors have shown that the phased array technique allows a significant enhancement of the testing performance in the bottom nozzle field. The enhancement does not reduce the sensitivity usually achieved for incipient cracking detection in the inner wall of the vessel bottom. Mechanical strength analyses indicate that crack formation is to be expected primarily at the inside surface, in the area of the control rod or instrumentation channel. (orig./DG).

2
(ECN-I–91-008)
Recalculations of HDR test T31.5 analyses with contain version 1.1. Velemà, E.J. (Netherlands Energy Research Foundation, Petten (Netherlands)). Jan 1991. 32p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603311.

This document presents the results of recalculations of the ISP-23 analysis and the additional hydrogen benchmark analysis. Both analyses were based on test T31.5 as performed in the HDR test facility in Germany. ECN participated in these analyses in the period 1988-1989. The new analyses have been performed to gain a better insight on how to model complex volume containment geometries with the CONTAIN computer program. Two recalculations have been performed: 1. a recalculation with a modified version of the 14 cell input model as has been used in the original calculation; 2. a recalculation with a more detailed input model. This latter model will also serve as basis for the upcoming ISP-29. (author). 8 refs.; 31 figs.; 1 tab.

3
(EPRI-NP–7520)
Radiation-field buildup at Monticello BWR with hydrogen water chemistry. Asay, D. (Electric Power Research Inst., Palo Alto, CA (United States); Radiological and Chemical Technology, Inc., Santa Clara, CA (United States)). Sep 1991. 88p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Primary recirculation piping at Monticello Nuclear Generating Plant was replaced in 1984. In November 1988, approximately midway through the third fuel cycle after pipe replacement, Monticello initiated hydrogen addition to mitigate the possibility of stress corrosion cracking. Due to expectedly high activity levels of Cobalt-60 observed in April of 1989, Northern States Power elected to decontaminate the entire primary system during the August 1989 refueling outage at the end of the third fuel cycle after pipe replacement. At the beginning of the fourth fuel cycle after pipe replacement, Monticello also initiated GEZIP in an effort to reduce future radiation level build-up. Four isotopic measurement campaigns of the Monticello recirculation system were performed since the system was decontaminated in 1989. These measurement campaigns and the radiation build-up at MNGP during fuel cycle 14 are the subject of this report. Northern States Power Company also performed a primary system decontamination at the end of cycle 14. the decontamination process was also monitored and the final report documenting these measurements, RCT-9107-2403, is appended to this report. 4 refs., 3 figs., 9 tabs.

4
(IWGFPT–36, pp. 102-109)
Study on fission gas release from high burnup fuel. Koizumi, S.; Umehara, H.; Wakashima, Y. (Toshiba Corp., Yokohama (Japan)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Detailed PIE has been carried out on three fuel assemblies irradiated in BWRs to 30 - 35 GWd/t. Our data base for fission gas release (FGR), has been extended to 39 GWd/t in rod burnup and to 43 GWd/t in pellet burnup. The FGR rate data measured on the three assemblies showed large scattering from 0% to 25%, as the previously reported data of less than 30 GWd/t. These scattering data can be related with the maximum powers they experienced beyond 10 GWd/t. The FGR rate seems to depend mainly on maximum powers, however, only a little on burnups. Micro gamma scanning and EPMA results revealed that local FGR rate varied radially in the pellet for a high FGR rate rod, i.e. almost 0% at the outer region and almost 100% at the center region. There existed a narrow transient band between them and local FGR rate showed a rapid change there. Ceramography and SEM observations showed that the local FGR rate variations related closely to pellet micro structural changes. At the center region, many large pores were found on the grain boundaries, connecting to each other and providing tunnels for gas release. The tunnel formation may do a key role to increase the local FGR rate. No remarkable changes were found at the outer region, except a thin pellet outer surface layer (pellet rim). The micro structure of the transient band indicated that the process of pore growth and tunnel formation related with the local FGR rate increase. The tunnels seem to control local FGR rate since their occurrence. Their effect, especially historical effect on FGR, should be considered in fuel behaviour analysis codes. The pellet rim showed to have very different structure. The original structure disappeared and very fine pores appeared. The observed rim structure looked still retaining most of fission gas. However, the numerical density of the pores is so high that pore connections and tunnel formations may occur and induce additional FGR. (author). 7 refs., 6 figs., 2 tabs.

5
(IWGFPT–36, pp. 125-131)
Application of ion microprobe analysis to the post-irradiation examination of BWR fuels. Imamura, M.; Ohuchi, A.; Ogata, K. (Nippon Nuclear Fuel Development Co. Ltd., Oarai, Ibaraki (Japan)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Ion microprobe analyses were conducted on UO_2 and UO_2-Gd_2O_3 fuels to investigate the irradiation behaviours of fission products and burnable absorber of gadolinium. The fuel rods were irradiated during 1 to 5 cycles in a commercial BWR, the Fukushima Daiichi No. 3 Reactor of the Tokyo Electric Power Co., Inc. The sample burnups ranged from 1 to 42 GWd/tU. The radial distribution of fission products, fissile material (Pu) and burnable absorber (Gd) were measured using a shielded ion microprobe analyzer. (author). 4 refs, 8 figs, 1 tab.

6
(IWGFPT–36, pp. 132-139)
Fission product distribution at different power levels. Lysell, G.; Schrire, D. (Studsvik Nuclear, Nykoeping (Sweden)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Two identical short BWR rods with a burnup of about 35 MWd/kg U were studied using a number of techniques. One of the rods was subjected to a slow power ramp ("bump") in Studsvik's R2 reactor, to a maximum linear heat rating of 43 kW/m, with a steep power profile. The other rod had experienced a maximum power of 26 kW/m during its final cycle, and was used as a reference. Various gamma measurements were performed immediately after the bump, enabling quantitative measurement of short-lived isotopes such as I-131, Te-132 and Xe-133 on both fuel and cladding specimens. These measurements indicated that a surplus of iodine over cesium could reach the cladding at pellet interfaces in a ramp situation. Diametral gamma scanning, EPMA, quantitative microscopy and density measurements were performed on fuel samples from two local power positions from the bumped rod, and from the reference rod. The gamma and EPMA measurements established the radial release/redistribution of xenon and cesium in the fuel. The quantitative microscopy and density measurements characterized a number of microstructural changes in the fuel, such as the local swelling/densification, grain growth, grain edge porosity fraction and intergranular porosity surface-to-volume ratio. The fuel microstructure was compared to the xenon fission gas release. (author). 11 refs, 8 figs, 4 tabs.

7
(IWGFPT–36, pp. 140-146)
In-reactor thermo-mechanical measurements on LWR fuel rods in the high burnup range. Kolstad, E.; Devold, H.; Tempest, P.; Loesoenen, P. (Institutt for Energiteknikk, Halden (Norway). OECD Halden Reaktor Projekt). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

The extension of fuel burn-up beyond previously accepted levels is currently being applied in varying degrees throughout the nuclear industry, with the aim of improving fuel economics and reducing the spent fuel volume. So it is necessary that the current fuel knowledge base should be extended. Modifications of fuel rod/assembly concepts, together with fuel management schemes, should be gradually implemented so that the operation of power reactors becomes even more reliable and flexible than it is today. Extrapolation to extended burn-up levels does not cause concern but will have to be made in steps, in order to demonstrate expected performance trends. The fuel testing programmes at the OECD Halden Reactor Project have over the years significantly contributed to the understanding of LWR fuel behaviour in the high burn-up range. A broad range of versatile and integrated in-reactor test rigs and high pressure loops have been developed which allow simulations of LWR irradiation conditions, comparative testing of alternative fuel rod designs and use of test segments pre-irradiated in power reactors. A number of in-core instruments and experimental techniques have been developed for detailed investigations of various aspects related to the thermal behaviour, fission product release and mechanical response of high burn-up LWR fuel rods, under a variety of operating conditions. The paper reviews recent measurements in the area of burnup-dependent steady-state and transient thermal behaviour of fuel rods, intermixing of fission and helium filler gases in the pellet cladding gap, fission gas release kinetics under changing heat loads and power excursions (burst release) and dimensional changes of fuel rods subjected to cyclic load changes. (author). 14 refs, 12 figs.

8
(IWGFPT–36, pp. 147-159)
Improved PCI and FGR performance of LWR fuel using rifled cladding. Mogard, H.; Kjaer Pedersen, N. (Studsvik Nuclear, Nykoeping (Sweden)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

A long term in-house R and D program, structured to test and evaluate the "rifled" clad fuel concept has been successfully concluded at Studsvik. The objective of the program was to modify and optimize the fuel pellet/clad gap configuration to attain improved fuel performance within critical areas like pellet-cladding interaction (PCI) and fission gas release (FGR). The approach chosen was to modify the pellet/clad gap configuration by modestly undulating, "rifling", the bore of the cladding tube. The program was concluded by a number of irradiation tests at the burnup level of 20 MWd/kgU using BWR 8x8 type fuel. The objective of the program has been successfully fulfilled. An improved PCI failure resistance has been attained, but not, however, precisely quantified. Also substantial improvements in the FGR behavior under varying operational conditions have been indicated. Fuel code analyses support the experimental findings. Local enhancement of gap fission gas concentration due to athermal gas release accumulating at high burnup, or due to stochastic flow restrictions at low and intermediate burnup (BWRs in particular) is expected to be reduced by the rifled cladding design, which will lead to reduce overall fission gas release. The fuel design also favours cladding stress equalization due to the ability of the fuel-cladding interface to suppress high localised tangential forces. This behavior tends to suppress the PCI failure initiation and progression. The post failure ingress of water/steam will be rapid and inhibits further PCI/SCC (stress corrosion cracking) initiation and propagation. As a side issue of the program the remedy effect of graphite coated pellets was investigated. No failures were experienced under the extreme ramp tests conditions applied and the accompanying FGR became modest. A mechanistic explanation of the improved failure performance is suggested. (author). 17 refs, 12 figs, 1 tab.

9
(IWGFPT–36, pp. 180-182)
OKG policy to avoid fuel failures. Ekbom, L.R.; Lundin, P.; Wiksell, G.; Jonsson, A. (OKG AB, Figeholm (Sweden)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

The operation of the core should be performed in such way that fuel failures can be avoided. If a fuel failure still arises, the knowledge and competence should be on such a high level that fast and accurate corrective actions can be worked out. To fulfill this policy, OKG Core and Fuel Management Units have evaluated procedures, which will here be presented. The personnel consist of a balanced mix of university educated engineers and engineers recruited from the control room. All core and fuel management personnel are located at the site, resulting in a closer contact to the operational personnel. The fuel design is very carefully examined already in the bid evaluation, and new designs are evaluated with lead test assemblies. Certain assemblies in every reload batch are followed during the entire in-core operation time. Every fuel failure is evaluated, so that the mechanism behind that failure is revealed and understood. The refuelling and operation as well as physics fuel design for all units is based on own calculations. 2D analysis and cross sections from CASMO (Studsvik) and 3D power distribution from POLCA (ABB Atom) are used together with own programs as engineering tools. In all three reactors the "double diagonal" and "single diagonal" patterns are used for refuelling, resulting in relatively few fuel shuffling. The design for the enrichment distribution and burnable absorber of the fuel and the refuelling of the core is optimized to make it possible to use monosequence operation. Regular meetings with other utilities, both Swedish and foreign, together with participation in international cooperation programs and conferences ensure that experience exchange exist. A very good and close contact have also been established with the fuel vendors. All this together gives a good platform for a non-leaking philosophy. (author).

10
(IWGFPT–36, pp. 182-190)
Reduction of effective thermal conductivity in high burnup fuels. Kitajima, S.; Matsumura, T.; Kinoshita, M. (Central Research Inst. of Electric Power Industry, Tokyo (Japan)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

We have developed a fuel performance evaluation code, EIMUS, improving the FEMAXI-III code in 1984. Recently modifications were made in this code to analyze high burn-up fuels. It was found that, as burn-up increased, pellet temperatures increased for the same liner heat rate in temperature measurements. Temperature increments are resulted from the reduction of effective thermal conductivity. It may be results of the following mechanisms: 1. The occurrence of many micro-cracks in pellets; 2. Reduction of thermal conductivity UO2 matrix. Firstly, pellet temperatures increase due to the micro-cracks because of solid-gas temperature slips at the crack gaps. Secondly the thermal conductivity of UO2 matrix reduces with O/U increment at high burn-up. We have made modeling of these phenomena to describe the reduction of the effective thermal conductivity and investigated effects of modeling parameters. This paper will also present comparisons between calculated, utilizing the developed model, and measured fission product gas releasing in experiments of RISO project phase 2. (author). 14 refs, 11 figs, 2 tabs.

11
(ORNL/tr–91/26)
Materials in the primary system of water-cooled nuclear power plants. Debray, W.; Stieding, L. (Oak Ridge National Lab., TN (United States)). 1991. Contract AC05-84OR21400. 34p. Translation of Lecture No. 3 delivered at the Power Conference, Lausanne, Switzerland, 1972, 25 pp. (CONF-720578–1). OSTI; NTIS; INIS; GPO Dep. Order Number DE92000158.

From International nickel power conference; Lausanne (Switzerland) (May 1972).

The first part of this document deals with the various phenomena of selective corrosion on austenitic CrNi steels, Inconel 600 and Incoloy 800 under the different conditions of light water reactor operation. The paper focuses particularly on the phenomenon of intergranular stress-corrosion cracking. For this purpose the results of the authors' own corrosion tests are presented and discussed in connection with recent results obtained at other research centers and with operating experience accumulated with boiling water reactors and pressurized water reactors. The switch in steam generator tube material from Inconel 600 to Incoloy 800 is described. In the second part of the paper the problems of cracking during the processing of low-alloyed steel are discussed. The motivation for this analysis was the worldwide occurrence of so-called underclad cracks in the steels ASTM A 508 cl. 2 (22 NiMoCr 37) and ASTM A 533 grade B class 1. It is shown that what is involved here is a specialized form of stress relief cracking, for which this type of steel has a certain sensitivity. Several crack tests of this type are described. It is proposed that the concept of relaxation embrittlement be introduced as the cause of these cracks. More detailed information is presented about this problem. 31 refs., 28 figs., 1 tab.

12
(TK–16-226-nr.9)
The performance of nuclear power plants. (Tweede Kamer der Staten-Generaal, The Hague (Netherlands)). Jul 1989. 11p. (In Dutch). OSTI; NTIS (US Sales Only); INIS. Order Number DE92605468.

Proceedings of the Second Chamber of the States-General(i.e. Dutch House of Representatives), Assembly-year 1988-1989; ISSN 0921-7371.

A survey is presented of failures in the Dutch nuclear power plants Borssele (10) and Dodewaard (5) reported during the year 1988. This reporting takes place, since 1987, on the basis of the international failure-reporting system. This system is based on the 'Incident Reporting System' of the IAEA. During 1988 no failures did occur which made particular safety measurements necessary. Also these failures did not have any consequence for the environment. During all failures the reactor safety system of both power plants did operate well. (H.W.).

13
(Windscale-Trans–1196)
Decommissioning of nuclear installations: study of underwater remote-control and separating techniques for the dismantling and disassembly of active RPV internals in KRB-A. (AEA Technology, Windscale (United Kingdom)). Jun 1990. 22p. Translated from German. Available from AEA Technology, Dounreay KW14TTZ.

Translated from German.

Planning work is described for the dismantling of the reactor internal fittings and the pressure vessel from a German nuclear power station, KRB-A. An inventory of the conditions in the reactor was carried out. A literature survey was undertaken to determine the state of the art in separating and handling techniques for the dismantling of nuclear installations. The optimal techniques for the dismantling of the pressure vessel and the internal fittings from the reactor are to be determined with the aid of a Working Group made up from several industrial companies. Several dismantling concepts are being developed separately and in some cases different priorities are deliberately being set. The results of the individual concepts will subsequently be compared and assessed with regard to their possible application in this reactor and an optimal concept will then be developed from all the proposals. Inactive preliminary experiments on the disassembly of the KRB-A steam drier have also been planned and the necessary preparations made. (author).

14
Operating experience with nuclear power plants in Germany. Dittmar, H.; Gutena, J. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 71-93 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (In German). (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

This lecture deals with the operating experiences gained from 21 German PWRs and BWRs. Examples of the most important performance indicators are given. The essential causes of failures and shut-downs are dealt with and the reasoning behind planned outage are explained. Additionally radiation doses to service and maintenance personnal, emissions of radioactive air and water are reported. (orig.).

15
Requirement profiles qualification of materials for LWR primary loop components. Tenckhoff, E.; Erve, M. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 95-119 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (In German). (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

The paper reviews the materials testing and qualification work within the framework of the Siemens Materials Programme for LWRs. The materials testing, properties and operating performance are reported. The materials programme delivered an essential contribution to an enhancement of power plant reliability and safety, and thus to a reliable energy supply by nuclear power plants. (orig./HP).

16
GE's advanced boiling water reactor. Wilkins, D.; Redding, J.; Berglund, R.C.; Steiner, H. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 287-298 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

GE has two new, advanced and simplified Boiling Water Reactors (BWRs). The 1300 MWe Advanced BWR (ABWR) is under construction in Japan and is close to receiving VSNRC licensing approval. The 600 MWe Small/Simplified BWR (SBWR) continues to be developed and will soon be undergoing licensing review in the U.S. To successfully address the problems encountered by today's fleet of nuclear plants, both the ABWR and SBWR use the best proven features from BWR designs from Europe, Japan, and the United States and state-of-the-art electronics, computer, turbine and fuel technology. Further improvements in safety, performance, and economics are made by simplifying the designs and using so-called 'passive' features. Because the ABWR and SBWR share this technology, the only essential differences are in power rating, how core flow is recirculated, and the extent to which the safety systems rely on active or passive features. (orig.).

17
Europe's new reactors. Pedersen, Tor. *Physics World (United Kingdom);* 4: No. 8, 37-40 (Aug 1991).

Increasing global energy requirements have led to new developments in nuclear power generation. Advanced reactor designs being generated in Europe are described, and fall into two categories "evolutionary" or "innovative". The former are modest modifications to Light Water Reactors (LWRs), the latter, while incorporating LWR technology, emphasize passive safety features. (UK).

18
ABWR certification work brings US licensing stability nearer. Wilkins, D.R.; Quirk, J.F. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 51-52 (Jun 1991).

The Advanced Boiling Water Reactor (ABWR) is now approaching Final Design Approval by the US Nuclear Regulatory Commission (NRC) and will then proceed on to the certification phase of the NRC's new standard plant licensing process. Successful completion of this will usher in a new era of standardization and reactor licensing stability in the US. (author).

19
An assessment of eight void fraction models. Chexal, B.; Horowitz, J.; Lellouche, G.S. *Nuclear Engineering and Design (Netherlands);* 126: No. 1, 71-88 (Apr 1991).

This paper provides an assessment of eight well known models and correlations for predicting the void fraction. The void fraction predictions are compared using steady-state steam-water test data for vertical configurations that included almost 1500 data points representing several areas of interest to nuclear analysts such as: (1) high-pressure-high-flows, (2) high-pressure-low-flows, (3) low-pressure-low-flows, (4) countercurrent flooding limitation, (5) natural circulation flows, and (6) co-current drownlows. The data were representative of PWR and BWR fuel assemblies and pipes up to 18 inches in diameter. (orig.).

20
Analysis of multidimensional and countercurrent effects in a BWR loss-of-coolant accident. Shiralkar, B.S.; Dix, G.E.; Alamgir, M. *Nuclear Engineering and Design (Netherlands);* 126: No. 1, 127-136 (Apr 1991). (CONF-891004–).

From 4. international topical meeting on nuclear reactor thermal-hydraulics (NURETH-4); Karlsruhe (Germany) (10-13 Oct 1989).

The presence of parallel enclosed channels in a boiling water reactor (BWR) provides opportunities for multiple flow regimes in cocurrent and countercurrent flow under loss-of-coolant accident (LOCA) conditions. To address and understand these phenomena, an integrated experimental and analytical study has been conducted. The primary experimental facility was the steam sector test facility (SSFT), which simulated a full scale 30deg sector of a BWR/6 reactor vessel. Both steady-state separate effects tests an integral transients with vessel vlowdown and refill were performed. The presence of multidimensional and parallel-channel effects was found to be very beneficial to BWR LOCA performance. The best estimate TRAC-BWR computer code was extended as part of this study by incorporation of a phenomenological upper plenum mixing model. TRAC-BWR was applied to the analysis of these full scale experiments. Excellent predictions of phenomena and experimental trends were achieved. (orig.).

21
Sensitivity analysis of thermal-hydraulic parameters and probability estimation of boiling transition in a standard BWR/6. Aceil, S.M.; Edwards, D.R. *Nuclear Technology (United States);* 93: No. 2, 123-130 (Feb 1991).

This paper reports on the development of a model correlating the minimum critical power ratio (MCPR) of standardized boiling water reactors (BWRs) to a set of thermal-hydraulic variables in parameter space. A statistical approach with fractional factorial sampling along with response surface methodology and orthogonal central composite design are employed. The COBRA-II computer code, after modifications to include MCPR calculation, is used to simulate the thermal hydraulics of the BWR/6. The sensitivity is obtained by differentiating a quadratic equation that represents the state of the system with respect to its constituent variables. Taking this correlation as a joint multivariable probability distribution function and using crude Monte Carlo integration techniques, the probability of boiling transition during normal operation in a standardized BWR/6 is estimated. The result agrees very well with data available in open literature.

22
Nuclear energy system using pelletized fuel in a boiling liquid reactor. Schoessow, G.J. USA Patent 4,976,913/A/. 11 Dec 1990. Filed date 24 Apr 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This paper discusses a system for producing a superheated vapor by employing a nuclear fission reaction in a reactor containing pelletized fuel. The reactor being a closed pressure vessel partially filled with a reaction medium including a coolant liquid and the pelletized fuel dispersed therein, two or more nuclear reaction control rods adjustably positioned in the medium, means for continuously introducing pellets of fuel into the reaction medium and means for continuously removing pellets of spent fuel from the reaction medium; means for introducing the coolant liquid into the reactor, and means for removing super-heated vapor of the coolant liquid from the reaction medium. The pellets of fuel comprising substantially spherical pellets containing a core of porous graphite encased in a high melting point matrix having dispersed therein tiny pellets of a mixture of graphite and nuclear fissionable material, and a substantially spherical coating over the matrix of a metallic alloy having a melting point above about 4000°F.

23
BWR core flow measurement enhancements. O'Neil, T.J.; McGrady, J.A. (to General Electric Co., San Jose, CA (USA)). USA Patent 4,975,239/A/. 4 Dec 1990. Filed date 23 Jan 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes a boiling water nuclear reactor a process for estimating core flow rate. It comprises: providing a reactor core with a nuclear reaction therein generating neutrons having a core plate fluid flow barrier at the bottom portion of the reactor core; providing local power range monitors arrayed throughout the core for the measurement of moderated neutrons for the determination of the power of the nuclear reaction in the core; providing for coolant upflow through the core plate fluid flow barrier and the core for moderating neutrons from the nuclear reaction for the critical continuation of the nuclear reaction. The upflow being at the core plate fluid flow barrier at a single liquid phase portion of the core and continuing to a two phase steam/ coolant region of the core; measuring the pressure differential of the upflowing coolant across the core plate fluid flow barrier; measuring the power of the rector at the local z power range monitors; determining a quadratic relation for coolant flow between the local power range monitor readout and the pressure differential cross the core plate; and, computing flow from the quadratic equation.

Light-Water Moderated, Nonboiling Water Cooled

24
(ECN-I–91-012)
Description of the CONTAIN input model for the Borssele nuclear power plant. Velema, E.J. (Netherlands Energy Research Foundation, Petten (Netherlands)). Feb 1991. 20p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603312.

This report describes the standard CONTAIN input model for the Borssele Nuclear Power Plant (NPP) that has been developed by ECN. This standard model will serve as a basis for analysis of the thermohydraulic and severe accident phenomena inside the Borssele containment. Boundary conditions for specific containment analyses can easily be implemented in the input model. As a result ECN will also be able to react quickly on questions from the utilities or the authorities. The report also includes brief descriptions of the Borssele NPP and the CONTAIN computer program. (author). 6 refs.; 10 figs.; 3 tabs.

25
(EPRI-NP-7380)
Nuclear plant design and modification guidelines for PWR steam generator reliability. (Electric Power Research Inst., Palo Alto, CA (United States); GEBCO Engineering, Inc., Sebastopol, CA (United States)). Sep 1991. 310p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Operating experience gathered from PWR plant operation during the 1960's and 1970's has been incorporated into a series of design guidelines for secondary plant systems and steam generators. Specific guidelines included in this volume are: plant design for PWR steam generator inspection and nondestructive testing, revision 1; guidelines for design of steam generator blowdown systems, revision 1; plant design guidelines for layup and cleanup of steam, feedwater, and condensate systems, revision 1; design guidelines for plant secondary systems, revision 1 and plant design for steam generator replaceability, revision 1. The guidelines are intended to address those aspects of new plant design which will minimize corrosion damage to steam generators by controlling impurity ingress, facilitate steam generator nondestructive testing and provide for eventual replacement of steam generator if necessary. The guidelines, last revised in 1986, are primarily applicable to new plant construction, however, some of the guidelines may also be applicable to major backfits to existing plants.

26
(EPRI-NP-7396-M)
Effect of lithium hydroxide on primary water stress corrosion cracking of Alloy 600 tubing. Jacko, R. (Electric Power Research Inst., Palo Alto, CA (United States); Westinghouse Electric Corp., Pittsburgh, PA (United States)). Sep 1991. 11p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Primary water stress corrosion cracking (PWSCC) studies were performed on Alloy 600 in simulated PWR high lithium primary water. Tests were conducted at 330°C with Li concentrations ranging from 0.7 to 3.5 ppM in solutions containing boric acid and dissolved hydrogen. Highly stressed, Alloy 600 reverse U-bend specimens (RUBs) were predominantly used for tests. Both mill-annealed (MA) and thermally treated (TT) Alloy 600 were tested. The large number of specimens tested allowed the use of rigorous statistical techniques to interpret the variability of PWSCC performance. Results of tests of MA 600 RUBs at 2 stress levels show no effect of chemistry on the time to initiate PWSCC cracks over the range from 0.7 to 3.5 ppM Li. However, results for TT 600 RUBs and in MA 600 RUBs at a third stress level show the tendency for a shorter time to initiate PWSCC cracks at a Li concentration of 3.5 ppM. Analysis suggests that certain Alloy 600 components may experience an increase in PWSCC by using the higher Li content primary water due to a subtle influence of chemistry on PWSCC. 5 refs. 8 figs., 3 tabs.

27
(EPRI-NP-7494)
Hideout and return of complex mixtures in crevices. Balakrishnan, P.V. (Electric Power Research Inst., Palo Alto, CA (United States); Atomic Energy of Canada Ltd., Chalk River, ON (Canada)). Sep 1991. 50p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Hideout of mixtures of sodium chloride, sodium sulfate and calcium chloride were studied in crevices packed with carbon fiber or carbon powder packing, at a superheat of 90°F and heat fluxes varying from 27,280 to 86,282 Btu/ft^2.h (86 to 272 kW/m^2). Solute was found to accumulate at a linear rate unit the crevice approached saturation capacity. The hideout rate depended primarily on the heat flux and was given by the product of the concentration in the bulk water, the evaporation rate in the crevice and an efficiency factor. The efficiency for the fibre-packed crevice was very nearly 100%, while that for the powder-packed crevice was in the range 60 to 80%. No preferential hideout of one species over another was observed. Chemistry calculations using the code MULTEQ indicated that Na_2SO_4, $CaSO_4$ and NaCl were present in the crevice as solid and that the boiling point elevation of the solution in the crevice was about 45°F (25°C). The solute species were released when the heat flux in the crevice was turned off. The pH of the effluent went through a maximum value higher than that of the feed solution during the return phase; the crevice pH was probably high during the hideout phase and low during the return phase. The hideout process is examined in the light of an adaptation of the existing models. The hideout return is shown to be diffusion-controlled. 11 refs., 14 figs., 6 tabs.

28
(EPRI-NP-7499-M)
Hot cell examination of Oconee-2 fuel rods. Beauregard, R.J.; Mayer, J.T.; Pyecha, T.D.; Papazoglou, T.P. (Electric Power Research Inst., Palo Alto, CA (United States); Babcock and Wilcox Co., Lynchburg, VA (United States). Research and Development Div.). Sep 1991. 19p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Four non-failed fuel rods from Duke Power Company's Oconee-2 reactor were examined in the B&W hot cells. The purpose of the program was to determine the cause(s) of failure in "sister" rods located adjacent to the assembly instrumentation tube, but which were thought to be too badly damaged to provide useful information. The rods had operated at relatively high power levels during their first cycle, which appeared to be a contributing factor to the failures. Non-destructive examinations included visual and eddy-current examinations, gamma-scanning and rod growth measurements. Following rod puncture and plenum gas analysis, several rod sections were destructively examined using visual techniques, metallography and scanning electron microscopy. The major findings from the examination were regions of greater than expected cladding OD oxidation and hydriding in regions with highest cladding temperature. Highly localized oxidation was found nearest the fuel assembly central instrument tube. The extensive hydriding was associated with fuel column gaps which acted as low-temperature sinks for hydrogen diffusion. Based on the findings in the sound rods, two failure mechanisms were identified that could have resulted in cladding failure. These were rapid through-wall oxidation or massive hydriding due to local temperature troughs that acted to concentrate hydrogen generated by rapid corrosion. 2 refs., 3 figs., 5 tabs.

29
(EPRI-NP–7512-M)
PWR full-reactor coolant system decontamination. Aspden, R.G.; Grand, T.F. (Electric Power Research Inst., Palo Alto, CA (United States); Westinghouse Electric Corp., Pittsburgh, PA (United States). Energy Systems Business Unit). Sep 1991. 37p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

The overall objective of the current program is to identify and address all aspects of full system decontamination with the purpose of qualifying at least one process for PWR use. The objective of the current study is to provide baseline data on the performance of materials on the primary side after exposure to three cycles of the LOMI process. The technical feasibility of applying this process will be determined in a subsequent task. The general corrosion characteristics of over 39 materials were evaluated for some combinations of material, type of specimen (coupon and creviced coupons), and loop velocity (0, 5, 20 and about 150 ft/sec). At velocities of less than or equal to 20 ft/sec, sixteen types of specimens were employed to evaluate localized corrosion and stress corrosion cracking. Some of the specimens were examined after one cycle and all of the specimens after three cycles of exposure. Hybrid expansion joints were also evaluated after an additional, simulated steam generator exposure to beginning-of-life reactor coolant for 90 days. The results of these evaluations are presented in this report. 10 figs., 8 tabs.

30
(INIS-BR–2814)
Two-phase coolant pump model of pressurized light water nuclear reactors. Santos, G.A. dos; Freitas, R.L. (Coordenadoria de Projetos Especiais (COPESP), Sao Paulo, SP (Brazil)). (Associacao Brasileira de Ciencias Mecanicas, Rio de Janeiro, RJ (Brazil)). 1990. 5p. (In Portuguese). (CONF-9012116–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92605481.

From 3. National Meeting of Thermal Sciences; Itapema (Brazil) (10-12 Dec 1990).

The two-phase coolant pump model of pressurized light water nuclear reactors is an important point for the loss of primary coolant accident analysis. The homologous curves set up the complete performance of the pump and are input for accidents analysis thermal-hydraulic codes. This work propose a mathematical model able to predict the two-phase homologous curves where it was incorporated geometric and operational pump condition. The results were compared with the experimental tests data from literature and it has showed a good agreement. (author).

31
(IWGFPT–36)
Fuel performance at high burnup for water reactors. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Water Reactor Fuel Performance and Technology). Feb 1991. 216p. (CONF-9006387–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484.

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

The present meeting was scheduled by the International Atomic Energy Agency, upon proposal of the Members of the International Working Group on Water Reactor Fuel Performance and Technology. The purpose of this meeting was to review the "state-of-the-art" in the area of Fuel Performance at High Burnup for Water Reactors. Previous IAEA meetings on this topic were held in Mol in 1981 and 1984 and on related topics in Stockholm and Lyon in 1987. Fifty-five participants from 16 countries and two international organizations attended the meeting and 28 papers were presented and discussed. The papers were presented in five subsessions and during the meeting, working groups composed of the session chairmen and paper authors prepared the summary of each session with conclusions and recommendations for future work. A separate abstract was prepared for each of these papers. Refs, figs and tabs.

32
(IWGFPT–36, pp. 17-23)
Status report on the IAEA's WREBUS study. Lang, P.M.; Pazdera, F. (USDOE, Washington, DC (USA)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

In late 1987, the International Atomic Energy Agency initiated a multinational study called Water Reactor Extended Burn-Up Study, or WREBUS. The principal purpose of the study is to determine whether economic incentives may exist to extend the burnup of water reactor fuel beyond the range which has previously been accepted as economically justified for utility implementation; i.e., beyond discharge batch average values of about 45 MWd/kgU for BWRs and 50 MWd/kgU for PWRs. This study has been initiated by assuming that significantly higher burnups would not introduce serious new technical problems which would substantially increase costs, by determining required enrichments for the higher burnups for a variety of reactors and cycle lengths, and by computing corresponding fuel cycle costs first for an agreed-upon set of economic parameters and second (optionally) for any country's own set of economic parameters. Sensitivity studies were done to determine the importance of each of the principal economic parameters. A benchmark exercise was also conducted in which each country (or country team) used the same enrichments and reactor characteristics; this exercise showed that within round-off error each of the fuel cycle cost codes used by the various participants gave essentially the same results for the same inputs. Following the economic studies, the technical issues foreseen for burnup extension beyond the present ranges are reviewed, together with potential licensing, safety, and environmental effects. This review provides an indication of the extent to which the previous assumption of no serious new technical issues may be valid and indicates the scope and nature of research and development efforts that would be needed to proceed to the higher discharge burnups studied. The WREBUS study is till under way; it is the intent of this paper to provide a status report on its results. 6 figs, 2 tabs.

33
(IWGFPT–36, pp. 24-30)
Incentives for extended burnup for WWER reactors. Yakovlev, V.V.; Proselkov, V.N.; Pazdera, F.; Onufriev, V.D.; Bibilashvili, Yu.K.; Valvoda, Z.

(Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (USSR). Inst. Atomnoj Ehnergii). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

In the paper steps taken to increase burnup in WWER 440 and WWER 1000 reactors are discussed. Economic incentives to burnup extension above 45 MWd/kgU are also analyzed. The results of economic burnup optimization for WWER 440 with 12 and 18 month cycles and WWER 1000 with 12 month cycles up to 65 MWd/kgU, performed within the framework of IAEA WREBUS activity, are presented. Study of sensitivity to economic parameters is also mentioned. The performed analyses show reasonable incentives for burnup extension as high as 60 MWd/kgU. The optimum burnup is most sensitive to backend costs. Possible burnup limiting factors are also discussed. (author). 5 refs, 6 figs, 2 tabs.

34

(IWGFPT–36, pp. 33-38)

Fragema fuel behaviour at high burnups. Morel, M.; Gautier, B.; Combette, P. (Fragema, 69 - Lyon (France)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

In this paper, we shall be presenting some of the results acquired to date within the scope of the French program, describing for example the irradiation campaign in the Gravelines 5 reactor, in which two Fragema assemblies produced a total burnup approaching 58,000 Mwd/tU after 5 cycles. These assemblies containing 4.5% U235 enriched fuel rods reached this burnup with a power history representative of annual quarter-core fuel managements with cycle extension. The main results for the 4th and 5th irradiation cycles of these assemblies will be presented and compared with those acquired earlier either in experimental reactors (CAP and BR 3) or in power reactors. The results cover the following ares: Internal pressure and fission gas release; pellet/cladding interaction; fuel rod cladding strain; fuel rod growth; cladding waterside corrosion. These results will be used to draw overall conclusions and to identify limiting phenomena such as cladding waterside corrosion. The paper will then review the possible design or materials options which will enable these burnup limits to be extended so as to meet future utility requirements. 6 figs.

35

(IWGFPT–36, pp. 39-46)

ABB-CE experience with high burnup PWR fuel performance. Corsetti, L.V.; Smith, G.P. Jr. (Combustion Engineering, Inc., Windsor, CT (USA)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

In 1977, the U.S. Department of Energy (DOE) embarked upon a program to improve the uranium utilization in light water reactors. An early study sponsored by the DOE indicated that extended cycle length with a concomitant increase in discharge burnup provided the most effective means to improve fuel utilization and reduce fuel cycle costs. Different fuel management strategies have been developed to accomplish this objective. The degree to which a particular strategy is adopted, varies with the objectives of a particular plant operator. Beginning in the 1970's, ABB-CE began conducting a variety of lead fuel assembly programs in commercial pressurized water reactors to support the goal of improved fuel utilization. The results from these programs have indicated that no abrupt changes in fuel performance occur as a result of high burnup, extended cycle operation. No evidence of any change in overall fuel rod reliability has been observed. As a result of this successful experience, batch average burnups of 45 GWd/MtU with lead-rod average burnups from 50-55 GWd/MtU are now common design objectives. Observations in fuel performance areas considered important to high burnup operation are summarized in this paper. High burnup, fuel operating experience is also reviewed. (author). 10 refs, 8 figs, 3 tabs.

36

(IWGFPT–36, pp. 46-52)

High burnup fuel behavior studies at NUPEC. Oishi, M. (Nuclear Power Engineering Test Center, Tokyo (Japan)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

The Nuclear Power Engineering Test Center (NUPEC) has been carrying out fuel irradiation tests on Japanese commercial BWR and PWR fuels since 1976, in order to provide a systematic and thorough base data on the in-pile fuel behavior, under the sponsorship of the Ministry of International Trade and Industry. The providing tests on both BWR and PWR fuel assemblies have been conducted for 11 years, and confirmed that fuel design and manufacturing were appropriate and reliable. Verification tests on high performance fuel, which has a capability to resist load follow operation, are being carried out since 1981. The current NUPEC program focuses efforts on the verification of high burnup fuel. In the PWR program, irradiation for four assemblies designed for 48 GWd/t started last year, furthermore twelve LTAs which will introduce improved designs with a target burnup of 55 GWd/t are expected to be irradiated in the future. In the BWR program, the irradiation of eight LTAs designed for high burnup is in progress with a target burnup of 50 GWd/t. To obtain reference data for high burnup fuel design, post irradiation examination on three 8x8 assemblies, one assembly irradiated in Fukushima Daiichi No. 3 and two assemblies irradiated in Shimane No. 1, was finished by March 1990, and systematic knowledge on the in-pile fuel behavior was extended up to an assembly burnup of 35 GWd/t. The fuel assemblies and rods showed no apparent deformation and anomaly. Fission gas release rate indicated large scatter from 0% to approximately 25% and the scatter increased beyond 10 GWd/t. Oxide thickness increase tended to saturate at higher burnup, and maximum oxide thickness was 106 microns. Characteristic fuel behaviors for higher burnup, such as pellet clad bonding and pellet rim structure, were found in both Shimane and Fukushima rods. (author). 6 refs, 9 figs.

37
(IWGFPT–36, pp. 52-57)
Progress in understanding high burnup phenomena. Hallstadius, L.; Grapengiesser, B. (ABB Atom AB, Vaesteraas (Sweden)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

As a response to the incentive towards increasing discharge burnup of nuclear fuel, ABB Atom conducts an extensive high burnup verification and PIE program that goes hand in hand with the development of improved designs and materials. Actual and potential burnup limiting phenomena are investigated, and appropriate remedies are tested and demonstrated. Cladding corrosion in BWR and, in particular, modern high-temperature PWRs, is an important such phenomenon which is remedied by new and improved materials. An increased heat transfer are in BWR design, as in SVEA 96 compared to SVEA 64, is beneficial. The larger number of fuel rods of SVEA 96, providing lower linear heat rating, also leads to a reduced fission gas release and pressure build-up in the rod, as well as improved dryout performance. It furthermore allows for operation without PCI restriction, i.e., without the use of a clad liner. The performance of BWR fuel channels with respect to irradiation induced bow has gained considerable interest recently. SVEA fuel operating in a symmetrized lattice reveals an excellent bow behaviour. Reuse of channels is strongly detrimental with respect to bow. (author). 10 figs.

38
(IWGFPT–36, pp. 69-77)
High burnup LWR fuel experience. Goldstein, L.G.; Strasser, A.A. (Stoller (S.M.) Corp., Pleasantville, NY (USA)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Statistics will be presented related to high burnup fuel experience of both the U.S. and European LWR fuel vendors. The number of batches at various burnups for various vendors will be summarized above 30 GWD/T for BWRs and above 35 GWD/T for PWRs. Coolant or off-gas activity data will be reviewed in an effort to assess the successes and failures related to high burnup fuel. In addition, the LTAs exposed to high burnup will be summarized. Any burnup limiting phenomena identified in lead test assemblies (LTA) of full batches will be discussed. Peak pellet, bundle average and batch average burnups will be correlated from the available data. (author). 10 figs, 5 tabs.

39
(IWGFPT–36, pp. 78-84)
Experience from examinations of fuel rods irradiated to high burnups in Loviisa reactors. Moisio, J.; Teraesvirta, R.; Ranta Puska, K. (Imatran Voima Oy (IVO), Helsinki (Finland)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Destructive examinations on selected high burnup fuel rods, irradiated in the Loviisa reactors of Imatran Voima Oy (PWRs of the WWER type), have been carried out at Studsvik Nuclear AB's hot laboratories since 1982. Together with the pool-side examinations and recorded operation information they provide a qualified set of performance data from a burnup range of 32 to 48 MWd/kgU for WWER fuel featuring annular pellets and Zr1%Nb cladding. Emphasis in the examinations has been put on dimensional changes, rate of cladding oxidation, fission gas release, and other phenomena that may become limiting at high burnups. In the 14 punctured rods, the fission gas release fraction has been around 0.5%, inevitably due to the athermal processes only. Four of these rods and seven additional rods were non-destructively gamma scanned for rod plenum gas inventory. The two methods agree as to the low gas release. The measured rod average diameter decreases due to cladding creep ranged from 25 to 60 μm in the high burnup rods. The final gaps were generally closed and in parts of the rods fuel-to-cladding bonding was observed. Moderate rod length increases were recorded. Only minimal cladding external corrosion of 1 to 4 μm was seen. The results of calculations by Technical Research Centre of Finland (VTT) were compared with the measured dimensional changes and fission gas release. It is concluded that the low maximum linear power, not exceeding 300 W/cm, and relatively early gap closure keep the fuel temperature below the gas release threshold. (author). 6 refs, 8 figs, 3 tabs.

40
(IWGFPT–36, pp. 85-93)
Post-irradiation examination results of high burnup demonstration fuel. Matsuoka, Y.; Toba, M.; Mori, K. (Kansai Electric Power Co., Inc., Osaka (Japan)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Japanese utilities are making preparations to increase the authorized fuel discharge burnup limit from its present value of 39 GWd/t (Assembly) to 48 GWd/t. As part of these preparations, five PWR utilities and two fuel vendors started a study to verify high burnup fuel performance, including a demo fuel irradiation program. Four fuel assemblies, two from Mitsubishi Heavy Industries (MHI) and two form Nuclear Fuel Industries (NFI), were selected for the demonstration. They were irradiated a 4th cycle at Ohi Unit 1 starting in 1986. The two NFI fuel assemblies selected had been irradiated since 1982. Their 4th cycle of irradiation produced a maximum assembly burnup of 45 GWd/t, with some rods approaching 48GWd/t. A sipping test after the 4th cycle of irradiation confirmed no leaking fuel rods. Dimensional measurements and visual inspections performed with an underwater TV after each cycle found no anomaly related to high burnup. Twelve fuel rods from one of two demo assemblies were selected to cover the range of burnup and various locations in the assembly. These rods were transported to a hot cell facility at JAERI in April 1989, and Post Irradiation Examination (PIE) work was started. Non-destructive testing (NDT) has been completed on three rods which are planned to be destructively tested. Destructive testing (DT) of the one rod which attained the maximum burnup is nearly completed, and the results are now under evaluation. NDT included visual inspection, profile measurement, gamma scanning, X-ray radiograph, and crud analysis. DT included puncture test, metallo- and ceramography, micro gamma scanning, pellet density measurement, plenum spring inspection, and fretting wear measurement. No anomaly was found in these rods. The irradiation of high burnup fuel on a commercial scale in Japan started in January 1990, at Takahama Unit 3. (author). 9 refs, 10 figs, 5 tabs.

41
(IWGFPT–36, pp. 94-102)
High burnup fuel rod behaviour and cladding mechanical properties under load follow conditions. Cappelaere, C.; Trotabas, M.; Cuavin, R.; Royer, J. (CEA, Saclay, Gif-sur-Yvette (France). Service des Elements Combustibles et Structures). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Twelve 17x17 Assemblies of CEA design were introduced in the first core of the CAP reactor (Cadarache-FRANCE). Five of them were irradiated during three cycles under PWR-load follow and frequently control simulated conditions. After that, twelve rods were extracted and examined in hot cells. Usual PIE were performed as well as mechanical tests in order to exhibit, if they exist, the effect of a numerous power cycling regime on the fuel behaviour and the cladding properties. 10 figs, 2 tabs.

42
(IWGFPT–36, pp. 110-116)
UO_2 irradiated at extended burnup: Fission gas release and correlated structural features. Coquerelle, M.; Walker, C.T. (Commission of the European Communities, Karlsruhe (Germany, F.R.). European Inst. for Transuranium Elements). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

UO_2 fuel irradiated over 1 to 4 reactor cycles and at average linear power ranging from 35.8 to 20.2 kWm^{-1} have been investigated in the α, γ Laboratory at the Institute for Transuranium Elements. The aim of this study was to determine the influence of increasing burn-up on fission gas release and the relationship between gas release and the fuel microstructure. Emphasis was placed on the investigation of fuel restructuring, and examination techniques such as quantitative optical microscopy, electron scanning microscopy and electron probe microanalysis have been used. Fission gas release measurements showed a trend towards increasing release as burn-up increased. Further, the investigation of the fuel microstructure established that fission gas release is strongly temperature sensitive, although there is very likely a minor athermal contribution from the outer rim of the fuel pellet. Fuel restructuring in the central part of the pellet was the cause for most of the gas release. (author). 4 refs, 7 figs, 4 tabs.

43
(IWGFPT–36, pp. 119-125)
Improvement of fuel pellet for high burnup. Doi, S.; Abeta, S.; Harada, Y. (Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Japanese PWR fuel have achieved the satisfactory performance within the assembly burnup of 39 GWd/t and is now going to take the next step toward the high burnup usage up to the 48 GWd/t. This high burnup is actualized without any improvements of the materials but with a reconsideration of the design methods. Moreover, utilities desire the higher burnup fuel up to the burnup of 55 GWd/t. In this case, one of the problems which should be solved is the potential increase of a fuel rod internal pressure due to the fission gas (f.g.) release from pellets. Then the fuel pellet with a large grain size is considered to reduce the f.g. release and two ideas, the additive to pellets and the improvement of the urania powder characteristics have been investigated to reach that. The former one is the pellet added with the niobia and was irradiated in BR3 reactor to the pellet peak burnup of 18 GWd/t. The results indicated the significant reduction of f.g. release. Another niobia added fuel is now irradiated to higher burnup. The latter idea is the pellet with the large grain size fabricated under the usual hydrogen sintering condition using the so called "active urania powder" which has the large specific surface area. At present, the pellet with the grain size of more than 30 μm has already successfully obtained with this powder. And both of these two types of the pellets with the large grain size plan to be irradiated in a domestic plant to demonstrate the f.g. reduction. (author). 1 ref., 8 figs.

44
(IWGFPT–36, pp. 159-163)
PWR waterside corrosion studies at Halden. Karlsen, T.M.; Takasaki, A.; Gunnerud, P.; Fjellestad, K. (Institutt for Energiteknikk, Halden (Norway). OECD Halden Reaktor Projekt). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Investigation of the effects of both thermal-hydraulic conditions and high pH on the corrosion rate of Zircaloy-4 are being conducted at the Halden Project by means of an in-pile PWR loop, able to produce well-defined and representative water chemistry and thermal-hydraulic conditions. In a first experiment, the effects of power cycling between 50 and 100% power and between one-phase cooling and surface boiling conditions were assessed. LWR segments pre-irradiated in a Siemens reactor were used for this test. After 60 power cycles, oxide layer thicknesses were measured and compared with the oxide layer of rods operated at constant power. No apparent corrosion enhancement due to the power/cooling cycling was found and the measured oxide layer growth was consistent with calculated thickness values. In order to determine the effects of high pH on corrosion rate, the loop is currently being operated with 4-4.5 ppm lithium (and 1000 ppm boron) in the coolant water. The fuel rods are being subjected either to one-phase cooling or to surface boiling conditions. After 80 full power days in the test reactor the first interim oxide layer measurements of the rods were performed. No evidence of increased corrosion rates as a result of the high lithium concentrations has emerged thus far. (author). 3 refs, 3 figs.

45
(IWGFPT–36, pp. 167-172)
Computer and experimental WWER fuel rod modelling for extended burnup. Strijov, P.N.; Dubrovin, K.P.; Yakovlev, V.V.; Pazdera, F. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (USSR). Inst. Atomnoj Ehnergii). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

To improve the economy, four-year instead of three-year refueling cycle has been started with WWER-440, and similar steps with WWER-1000 (three-year instead of two-year) are taken, both leading to considerable burnup extension. Fuel rod behavior of this reactors with extended burnup was analyzed with the computer code PIN-micro developed in cooperation between Soviet and Czechoslovak specialists. With the aim of further verification, calculations with this code were compared with results of irradiation of instrumented experimental fuel rods and postirradiation examination (PIE) performed at IAE Moscow. As an example some comparisons with PIE of WWER rods irradiated to burnup 60 MWd/kgU are shown as well as comparison with two rods with inside gas pressure instrumentation irradiated to 32 MWd/kgU. The results of variant calculations by the PIN-micro code are presented for a WWER-440 and WWER-1000 fuel rods for a four- and three-year refueling cycle, respectively. To optimize the initial internal gas pressure in WWER-440 fuel rods for four-year refueling cycle some variant calculations were performed whose results are also included in the paper. It is shown how the calculations performed allowed improvements in the fuel rod design. (author). 5 refs, 7 figs.

46
(IWGFPT–36, pp. 205-207)
Factors affecting WWER type fuel behaviour at extended burnup. Bibilashvili, Yu.K.; Novikov, V.V.; Pimenov, Yu.V.; Proselkov, V.N. (Vsesoyuznyj Nauchno-Issledovatel'skij Inst. Neorganicheskikh Materialov, Moscow (USSR)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

WWER-1000 fuel has achieved a satisfactory level of operational reliability within burnup of $\approx$ 43 MWd/kg U. It allowed the realization of the next stage of R and D and to achieve the burnup of $\approx$ 55 MWd/kg U using burnable absorbers. Results of experimental studies include the possibility of operation of fuel to higher burnup. (author). 2 refs.

47
(IWGFPT–36, pp. 208-213)
Burnup limits in PWR and in BWR fuel. Goldstein, L.G.; Strasser, A.A.; Sunderland, D.J. (Stoller (S.M.) Corp., Pleasantville, NY (USA)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

The design of future cores will be based on burnup limits that are currently expanding. The performance parameters that have limited the burnup potential for current fuels in BWRs and in PWRs will be identified. Any changes that are justified through data from lead test assemblies (LTAs) or other improvements will be reported. Performance issues such as Zircaloy growth, pellet-cladding interaction (PCI), corrosion, Zircaloy ductility, fuel rod internal pressure will be discussed relative to potential limits. Some of these are limiting because of fuel rod integrity and others because of licensing or commercial considerations. Potential limits that arise from nuclear design considerations will also be described for general cases where peaking or other limits may be reached at lower values than those due to mechanical integrity, for example. Where representative data are available for classes of BWRs and PWRs, operating limits will be projected for use by those utilities with plants in these categories. (author). 10 figs.

48
(ZJE–286)
Experience with the WIMS computer code at Skoda Plzen. Vacek, J.; Mikolas, P. (Skoda, Plzen (Czechoslovakia)). 1991. 74p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603321.

Validation of the program for neutronics analysis is described. Computational results are compared with results of experiments on critical assemblies and with results of other codes for different types of lattices. Included are the results for lattices containing Gd as burnable absorber. With minor exceptions, the results of benchmarking were quite satisfactory and justified the inclusion of WIMS in the production system of codes for WWER analysis. The first practical application was the adjustment of the WWER-440 few-group diffusion constants library of the three-dimensional diffusion code MOBY-DICK, which led to a remarkable improvement of results for operational states. Then a new library for the analysis of WWER-440 start-up was generated and tested and at present a new library for the analysis of WWER-440 operational states is being tested. Preparation of the library for WWER-1000 is in progress. (author). 19 refs.

49
French-German cooperation for the development of the next generation of PWR type reactor. Ruess, F. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 171-182 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (In German). (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

In April 1989, the two major reactor manufacturers in Europe, Siemens/KWU and Framatome, signed a contract for long-term cooperation on a broad basis. For this purpose, the common subsidiary 'Nuclear Power International', (NPI), was established. The contractual goals and the tasks of NPI are explained, referring primarily to the plans for a joint venture for the development of a future-oriented, French-German PWR technology. (orig.).

50
Design improvement to facilitate decommissioning of P.W.R. Dubourg, M. pp. 132-135 of Decommissioning and demolition 1990. Whyte, I.L. (ed.). London (UK); Thomas Telford Ltd. (1990). 254p. (CONF-9004114–).

From 2. international conference on decommissioning offshore, onshore demolition and nuclear works; Manchester (United Kingdom) (24-26 Apr 1990).

Organised by Institution of Civil Engineers, North Western Association; University of Manchester Institute of Science and Technology; University of Manchester.

By analyzing the design of both 900, 1300 MWe and future 1450 MWe PWR units, it appears that design improvements made in order to reduce occupational dose exposure of maintenance personnel and the development of automated tools for performing maintenance and repairs of major components, will contribute to the facility of future dismantling and decommissioning operations. (author).

51
Pressurized water reactors. Bruschi, H.; Mink, F.J. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 269-286 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

Two major Westinghouse design programs have culminated in the safest and most economical nuclear plant designs in existence: the advanced pressurized water reactor (APWR) and the advanced passive design. The APWR program has produced a fully tested and validated 1300 MWe (net) four-loop design and a 1070 MWe (net) three-loop design, both ready for commercialization. The 1300 MWe APWR reference design has received U.S. NRC design approval in letter form; formal design approval is imminent. The advanced passive design meets the industry need for an advanced smaller plant. The Westinghouse AP600 design, which unites the best of proven nuclear power technology with natural safety features, is conceptually complete. Westinghouse is currently engaged in a Dollar 120 million program (U.S.) to perform sufficient design work for a Standard Safety Analysis Report, expected in mid-1992, and for final NRC Design Approval expected at the end of 1993. Final NRC design certification should come in 1994, with commercial operation possible before the end of this decade. Westinghouse is also heading a program to develop a 960 MWe (net) version of the AP600 in cooperation with Japanese organizations. (orig./HP).

52
Electropolishing gains acceptance. Raney, H. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 30-31 (Jul 1991).

Electropolishing is gaining acceptance in the nuclear industry, with several US facilities making use of the technique and EdF deciding to electropolish all its new Pressurized Water Reactor steam generator heads. The end result of the technique is a pure surface, free of burrs and sharp edges, that is highly passivated. This greatly reduces the tendency of the metal to accumulate radioactive particles. (author).

53
Modernizing the VVER: a Soviet perspective. Denisov, V.P.; Stekolnikov, V.V.; Voznesenskii, V.A. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 16, 18-19, 21-22 (Jul 1991).

The design and development of the VVER, the Soviet pressurized water reactor concept, dates back to the mid-1950s, with the first unit entering operation in 1964. Extensive efforts are now going into modernizing operational plants and developing designs for the future. (author).

54
The steam generator replacement comes of age. Hennicke, H. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 23-24, 26 (Jul 1991).

The steam generator replacements in Pressurized Water Reactors performed over the last decade indicate a trend towards improved techniques, shorter schedules and less total radiation exposure - they have matured from an exotic plant modification to an (almost) routine maintenance activity. In retrospect, the foresight of some utilities to design into their plants the capability to remove steam generators through the equipment hatch seems to have paid off. (author).

55
Single pass inspection for steam generator tubes. Lloyd, E.A.; Spooner, J.B. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 26-28 (Jul 1991).

Establishing the integrity of the several thousand tubes in a Pressurized Water Reactor steam generator is an exacting task, and a vast number of eddy current systems have been developed to help in this work. The National Nuclear Corporation's new probe aims to incorporate all the main features of the many techniques available, while generating all the essential data in a single pass. (author).

56
How PSA (probabilistic safety assessment) can benefit future reactor design. Ellia-Hervy, A.; Lange, D. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 36-37 (Jul 1991).

Framatome has found that probabilistic safety assessment (PSA) has brought significant improvements to existing nuclear power plants. For future reactors, PSA is being used as a design tool and a means of ensuring improved performance. (author).

57
Improving outages at Oconee. Tuckman, M. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 40-43 (Jul 1991).

In its early years the Oconee nuclear power plant had problems which affected its outages, and it was plagued with forced outages. Later outages were partly dictated by Three Mile Island backfits, and it is only in the last few years that the operator has gained control. The last six refuelings have decreased markedly in length: all were 41-45 days. The safety of the plant has increased as outages shorten, but the success of an outage cannot be measured just by its length -the work that is to be accomplished and the time taken must both be considered. Strategies for successful outage management are discussed. (author).

58
Maintenance at Tepco. Katayanagi, Hiroshi. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 45-47 (Jul 1991).

In Japan, the law specifies that a reactor system must be inspected every 12±1 months, and the turbine generator system must be inspected every 24±1 months, to maintain plant soundness. The annual inspection and maintenance programme is therefore based on long range planning for periodic inspection and maintenance. The programme also depends on the standard periodical inspection cycle of each item of equipment. Details of the programme are given. (author).

59
Standard plants, standard outages: the EdF approach. Miron, J.L. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 47-50 (Jul 1991).

At the end of 1990 Electricite de France had carried out a total of 350 PWR refuelling outages. Although the French units are standardized the routine of the outages are not all the same. The major influences on outages were: setting up new organizations to apply quality assurance regulations; improving systematic experience feedback; incorporating modifications in the outage schedules; assumilation of computerized maintenance management by the sites. (author).

60
Case study: Loviisa 2 refuelling outage 1990. Vuorenmaa, A.; Savikoski, A.; Vaittinen, A. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 51-52 (Jul 1991).

Long term maintenance at Loviisa is based on an eight-year cycle; every eighth year inspection of the pressure vessel cladding, welds and nozzles; every fourth year visual inspection inside the reactor vessel and partial ultrasonic inspections of the vessel welds; in other years simple refuelling outages. The length of these outages has been reduced to 50, 40 and 20 days, respectively, and huge efforts are made to match major equipment overhauls and replacements to this programme. The details of the outage planning are described. (author).

61
Review of Paks outage results 1990. Lukacs, P.; Zsoldos, F.; Kiss, Z. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 53-55 (Jul 1991).

The year 1990 was not the most successful from an outage point of view at the Paks Nuclear Power Plant in Hungary -there were one or two long delays. Work at unit 4 had a delay of 10 days because of an error made during assembling the reactor vessel. While the outage of unit 3 was running, a feedwater pipe hanger problem was discovered - several hangers were found displaced from the right position. A general inspection of the affected system was required and this took about 11 days. Information about each outage is presented on diagrams, making comparison easier. These diagrams give information about deviations from the outage plan, about work hours performed during outages, and about collective exposure. (author).

62
Outage lengths in the USA. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 43-45 (Jul 1991).

Outage lengths vary markedly, from plant to plant and from year to year, depending on the scope of work performed, so direct comparisons cannot be made. However, outage lengths are listed for individual US PWRs and BWRs to show the range of recent outage experience. Information on outage lengths was obtained from industry sources. Plots showing trends in refueling outage length for US PWRs and BWRs were obtained from the Utility Data Institute. (author).

63
Transport coefficients for laminar and turbulent flow through a four-cusp channel. Dutra, A.S.; Mendes, P.R.S.; Parise, J.A.R. *International Journal of Heat and Fluid Flow (United Kingdom);* 12: No. 2, 99-105 (Jun 1991).

A study was performed to determine entrance region and fully developed heat transfer coefficients for laminar and turbulent flow in a four-cusp channel, for a thermal boundary condition of uniform wall temperature. A numerical solution was presented for fully developed laminar flow, and an experimental study was reported for turbulent flow. In the experiments, systematic variations of the Reynolds number were made in the range 5300-30,300. The results show that the heat transfer coefficients for the four-cusp channel are much lower than the coefficients for the circular tube. (author).

64
Modernizing the VVER-440/230. Mink, F.J. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 32-33 (Jun 1991).

The modernization of the VVER-440/230s is not fundamentally different from backfit projects on older pressurized water reactors which Westinghouse has completed elsewhere. However, carrying out such programmes only makes sense if the plants are expected to continue operation for their projected life or beyond. This clearly requires some licensing and political stability; both are essential if investors in the upgrading project are to be found. (author).

65
Insights gained from PSAs of French 900MWe and 1300MWe units. Brisbois, J.; Lanore, J.-M.; Villemeur, A.; Berger, J.-P.; Guio, J.-M. de. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 40-41, 44-45 (Jun 1991).

The two probabilistic safety assessments of 900MWe and 1300MWe Pressurized Water Reactors (PWRs) recently completed in France constitute an important knowledge resource for the assessment of PWR safety. One innovative feature of this research programme, which yielded many valuable lessons, comes from the fact that plant shutdown state and long term post-accident conditions were fully taken into account. (author).

66
Bringing designer and analyst together in Sizewell B containment tests. George, P.T. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 45-46 (Jun 1991).

An international workshop on the results of the Sizewell B model containment test brought together a wealth of experience in the design, analysis, construction, testing and licensing of concrete containments. The workshop, reviewed here, showed that the experience of the designer and the analyst is of critical importance. (author).

67
A simplified Japanese PWR. Matsuoka, Tsuyoshi. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 47-48, 50-51 (Jun 1991).

Mitsubishi's new small Pressurized Water Reactor (PWR) concept - the

Mitsubishi Simplified PWR - has the innovative features of hybrid safety systems (an optimum combination of active and passive systems) and horizontal steam generators. It is intended to be suitable for service anywhere in the world, with larger versions possible by going to 3- or 4-loop arrangements. The main features of the design are described. (author).

68
PIUS on the move in Italy. Barabaschi, S. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 52-53 (Jun 1991).

It is almost ten years since ABB Atom began working on the PIUS reactor design, aiming to improve Lighter Water Reactor (LWR) safety characteristics with inherent and/or passive features based on simple natural laws such as gravity and thermohydraulics. Many versions have been studied, but recently efforts have been concentrated on a 600MWe unit with four-loops. Basically, PIUS is a passive, simplified PWR, based on well-established LWR technology. In conventional LWRs, severe accidents with major releases to the environment can occur following core meltdown. The main safety aim, therefore, is to protect the core and its fuel. In PIUS, this is accomplished by submerging the core in a large pool of borated water, and by ensuring that the cooling capability of this water is greater than the power of the core. The detailed design work for PIUS is progressing, with a final design verification experimental programme planned. An NRC licensability review will begin this year and the design certification process, which should aid any decision to construct the first plant, may well start in 1992/93. (author).

69
Skid-mounted submerged PWR pool filter uses in-to-out flow. Scowen, P.A.; Gabelgaard, K. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 56-57 (Jun 1991).

A new submerged Pressurized Water Reactor (PWR) pool filter system is described which has been developed using Pall glass fibre filter cartridges. This system is now in operation in 14 European nuclear power stations, and results show that a completely turbid fuel pool can be made clear within one hour. (author).

70
Premixing of steam explosions: A three-fluid model. Amarasooriya, W.H.; Theofanous, T.G. *Nuclear Engineering and Design (Netherlands);* 126: No. 1, 23-39 (Apr 1991).

A transient, two-dimensional, three-fluid model is proposed to address the problem of premixing of molten fuel and coolant (prior to a steam explosion) in the lower plenum of a pressurized water eactor (PWR). Predictions for a typical reactor case are compared with a previous work that ignored steam-water slip and the energy change in fuel. (orig.).

71
Triggering and propagation of steam explosions. Medhekar, S.; Abolfadl, M.; Theofanous, T.G. *Nuclear Engineering and Design (Netherlands);* 126: No. 1, 41-49 (Apr 1991).

The propagating thermal interaction between two interdispersed liquids of highly dissimilar temperatures and boiling points is considered in terms of a two-fluid/three-phase model. Of particular interest is the transient escalation from local, fine-scale, contact (trigger) to a highly energetic event, known also as a 'steam explosion'. The model is utilized to compute and study such interactions for several liquid-liquid pairs and to highlight the importance of vapor volume fraction in the mixture on the propagation characteristics and the resulting intensity of energy release. Qualitative comparisons with tentative experimental data are also attempted. (orig.).

72
Intrinsically Safe and Economical Reactor (ISER). Wakabayashi, Hiroaki; Yoshida, Tomoaki; Asahi, Yoshiro. *Nuclear Engineering and Design (Netherlands);* 126: No. 1, 89-103 (Apr 1991).

The Intrinsically Safe and Economical Reactor (ISER) is designed based on the principle of a process inherent ultimate safe reactor, PIUS, a so-called inherently safe reactor (ISR). ISER has been developed joingly by the members of the Kanagawa Institute of Technology, the University of Tokyo, the Japan Atomic Energy Research Institute (JAERI) and several industrial firms in Japan. This paper describes the requirements for the next generation of power reactor, the safety design philosophy of ISR and ISER, the controllability of ISER and the results of analyses of some of the design-based accidents (DBA) of ISER, namely station blackout, accidents in which the pressurizer relief valve becomes jammed and stuck in open position and tube breaks in the steam generator. It is concluded that the ISER can ensure a wide range of contraollabitily and fuel integrity for all the analysed DBAs. (orig.).

73
Removing silica by reverse osmosis. Glenn, J. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 441, 43 (Apr 1991).

Silica in itself is a relatively innocuous substance with no adverse effects on the environment of a nuclear reactor. However, when system operating temperatures and pressures are increased during power production, silica can combine with cations to form a zeolite layer on fuel assemblies. This creates a potential for localized hot spots on the fuel cladding, which could result in fuel failures. Fuel failures in turn may result in higher activity releases to the environment, increased volumes of radwaste, and greater potential for radioactive contamination. For these reasons, silica levels must be maintained at relatively low concentrations in all systems which handle water in contact with the fuel assemblies. Various technologies exist for removing silica from borated water systems. However, most of these also remove the boron, which can be costly to replace. US Ecology determined that the reverse osmosis (RO) technology would provide the services required in the most efficient manner. Preferential separation of various constituents of a fluid can be achieved by applying pressure across a membrane. Different constituents permeate the membrane at different at different rates, and membrane materials are developed and applied based on their characteristic permeability rates for species of interest. The overall separation capability of a membrane system is defined as the rejection. For this application, cellulose acetate hollow fibre membranes were chosen as they demonstrated the most consistent rejection of silica (over 85%) with minimum rejection of boron (around 15%) from the borated water system. (Author).

74
Design, fabrication and construction of PWR nuclear islands: Framatome experience and Daya Bay plant. Diehl, J. *Nuclear Power Engineering*

(Hedongli Gongcheng) (China); 11: No. 6, 42-49 (Dec 1990). (In Chinese).

Framatome is the most representative of companies for nuclear island. Its staff of up to 6000 has contributed to the commissioning of 57 pressurized water reactor (PWR) units up to the end of 1989. Eight others are in the course of completion. More recently, construction of the Daya Bay NPP has begun in the People's Republic of China, equipped with two Framatome 900MW$_e$ class nuclear islands (improved models of the 900MWe class units already built in France and abroad). The heavy nuclear components, manufactured by Framatome, are erected by a Framatome-Spie Batignolles consortium, with considerable participation by the Chinese Company No 23.

75
In-reactor structure analysis of 5 MW THR. Shi Yongchang; Wu Honglin; Zhang Zhensheng; Zhou Xiaoping; Chang Huajian. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 6, 88-90 (Dec 1990). (In Chinese).

The structures of 5 MW THR and its characteristic are briefed insure the reactor core always flooding under water in any conditions. Some features of the integrative arrangemnt, hydraulic control rod drive system and spent fuel storage in the pressure vessel are different from normal PWR and BWR.

76
Research on physical calculation methods for 5 MW THR. Luo Jingyu; Shan Wenzhi; Hu Yongming; Du Jie. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 36-44 (Oct 1990). (In Chinese).

The configuration of the fuel assembly in 5 MW THR with its cruci-form control rod, water gap, poison rod, assembly box and water cell and the small dimension of the core bring about high heterogeneity and strong leakage which makes the physical design quite difficult. An effective method is developed to solve this problem. The basic concepts are introduced briefly, the foundation of the physical models and their special treatments are given in detail. The method is validated against eigenvalues of a series of critical experiments. We specially designed and the measured power distribution reported in the literature. Over all agreement is observed to be satisfactory. The maximum error of Local Peaking Factor and eigenvalue is 6% and 0.83% respectively. 10 eigenvalues and 13 control rods worth of the 5 MW THR have been calculated and compared with 1:1 zero power experiment results. The maximum error of the eigenvalue is 0.9%. On the start-up phase a series paraments of the 5 MW THR was measured. These include eigenvalue k_{eff}, control rod worth, equal temperature coefficent of the reactivity and excess reactivity. All results are satisfactory.

77
Fuel assembly of 5 MW THR. Xu Yong; Zhang Zhensheng. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 85-89 (Oct 1990). (In Chinese).

Design principle, feature, structure, analysis and evaluation of fuel assembly for 5 MW THR are given.

78
The design of primary heat exchanger for 5 MW THR. Li Rizhu. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 90-92 (Oct 1990). (In Chinese).

The characteristic of primary heat exchanger for 5 MW THR is described. The heat transfer coefficient and resistance coefficient of primary side of primary heat exchanger are given.

79
Pressure vessel and containment for 5 MW THR. He Shuyan; Xiong Dunshi; Liu Junjie. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 93-96 (Oct 1990). (In Chinese).

The Design and analysis of the pressure vessel and containment were done according to correspending Chinese codes, standards and ASME code. The overall arangement and the structural features of both the pressure vessel and the containment are described. Their design, calculations, stress analysis and strength assessment are briefly explained as well.

80
Thermo-hydraulic design of 5 MW THR. Gao Zuying; Li Jincai; Qian Like. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 53-56, 69 (Oct 1990). (In Chinese).

The thermo-hydraulic design criteria, methods and features of 5 MW THR are given. Some important parameters for two operation conditions (pressure water and pressure water-slight boiling) are also given.

81
Simulated experimental study on start-up of 5 MW THR. Jiang Shengyao; Yao Meisheng; Bo Jinhai; Wu Shaorong; Han Bing; Zhang Youjie. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 61-64 (Oct 1990). (In Chinese).

'The effect of two-phase flow instabitity on the start-up of natural circulation low temperature heating reactor is described. It is suggested that the process of boiling start-up should consist two steps-the pressurized start-up and the transition from pressurized condition to boiling condition'.

82
Physical start-up of 5 MW THR. Luo Jingyu; Xu Kuan; Xu Xiaolin; Shan Wenzhi; Jing Xingqing; Bian Hui. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 32-35, 52 (Oct 1990). (In Chinese).

The characteristics and the results of the fuel loading, the first criticality and the test of adjusting the control rods for 5 MW THR are summariged. The extrapolation method and pratical steps are introduced which ensure the reactor safety. To reduced the space effect, the special measurement method is adopted during the fuel loading and the extrapolation criticality. The experiments show that these methods are effective.

83
The commissioning of the 5 MW nuclear district heating plant. Su Qingshan; Zhang Dafang; Lu Jinfa; Chen Hua; Zhang Yajun; Sun Shuanliang. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 24-31 (Oct 1990). (In Chinese).

The commissioning of the 5 MW nuclear District heating Plant of Tsinghua University is briefly introduced in three aspects, namely: the working out of the commissioning document, performance of the commissioning and some important tests.

84
The quality assurance of 5 MW THR. Bi Shuxun; Wu Xumo. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 19-20, 31 (Oct 1990). (In Chinese).

The quality assurance system at each stage (degign, manufacture, construction, commissioning and operation) for 5 MW THR is described.

85
5 MW test heating reactor. Wang Dazhong; Dong Duo; Ma Changwen; Lin Jiagui. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 8-14 (Oct 1990). (In Chinese).

The 5 MW Test heating reactor, including the purpose of the project, main design data and features, key technical issues, safety analysis, the results from the commissioning and operating of the reactor is briefly described. It has shown that the reactor is an ideal heat source of the city's district heating as its inherent safety features and higher operational reliability.

86
Temperature and stress calculations for the bottom part of the reactor containment of the Temelin nuclear power plant during its construction. Mejzlik, L. *Inzenyrske Stavby (Czechoslovakia);* 39: No. 6, 181-188 (Jun 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

Concreting of the bottom part of the containment was performed continuously in horizontal rings at a rate of about 10 cm/h. The amount of cement in the concrete was 500 kg/m^3. The time behavior of temperature in the containment wall was tentatively calculated. The heat of hydration evolves nonlinearly and causes the concrete temperature to rise to about 60 to 70 degC. Theoretical calculations and practical temperature measurements were carried out in a concrete test block. The results of calculation of local and global tensions, moduli of elasticity and strength values are shown in the form of plots. The concreting procedure in question is associated with an appreciable hazard of crack formation, particularly of circumferential ones, nearly along the whole outer wall height of the containment. The concreting of the bottom part of the containment was completed in June 1990. The temperatures measured agreed very well with the calculation, and very thin cracks were detected in the bottom part of the containment. (M.D.). 16 figs., 2 refs.

87
Development of a two-dimensional computer code for the prediction of two-phase heat transfer in an experimental light water reactor irradiation capsule. Kennedy, T.D.A.; McAllister, S.; Markgraf, J.; Ruyter, I.A. *Nuclear Engineer (Institution of Nuclear Engineers) (United Kingdom);* 30: No. 3, 165-172 (Jun 1991).

At the High Flux Reactor of the Joint Research Centre Petten Establishment, Light Water Reactor fuel rod irradiations are conducted using irradiation capsules specially designed to simulate Pressurized Water Reactor and Boiling Water Reactor operational conditions. To improve the understanding of conditions within these capsules a computer code has been developed. This code analyses the natural circulation, boiling and condensation processes by which heat is transferred from the fuel rod, through the heat transfer medium, to the capsule outer wall. It also takes account of two-dimensional heat diffusion within the fuel pin and the capsule structure, combining this with a treatment of the phenomena of single- and two-phase heat transfer and buoyancy driven flow which occur within the heat transfer medium itself. Buyoyancy driven flow within the capsule is modelled very simply, assuming a single recirculating convection cell. Heat transfer is calculated for natural convection and partial or fully developed sub-cooled boiling, depending upon the local conditions along the length of the fuel pin. It is felt that despite the simplifications employed the code provides a novel approach for handling two-phase heat transfer in conjunction with two-dimensional heat diffusion in the whole system, and that scope exists for its further development and wider applications. (Author).

88
Cladding and heat treatment of the Sizewell B PWR pressurizer. Jordan, K.C.; Mathers, E. *Nuclear Engineer (Institution of Nuclear Engineers) (United Kingdom);* 30: No. 3, 155-163 (Jun 1991).

NEI International Combustion Ltd were contracted to design and supply the pressurizer for the Sizewell B PWR. The vessel, 16.1m long, 2.35m in diameter and 107mm thick, is fabricated from ASTM A508 Cl3 low alloy steel. The internal surfaces of the vessel are clad totally with stainless steel. Specification requirements and certain aspects of the design have required specialized cladding techniques to be developed, and extensive pre-weld and post-weld heat treatments to be carried out. This Paper describes the cladding processes which were used to clad the vessel, in particular the use of a TIG hardened robot which was employed to clad the nozzle bores. Welding parameters and cladding sequences are detailed, together with the results of chemical analysis. Mention is also made of the production experience, duty cycles and success rates. The heat treatment activities are similarly described, with detail of the heat treatment requirements and the methods of achieving the necessary control. (Author).

89
WWER-440 reactor fuel loadings with reduced neutron leakage. Darilek, P.; Majercik, J.; Petenyi, V.; Stepanovic, N. *Jaderna Energie (Czechoslovakia);* 37: No. 5, 163-170 (May 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

Fuel loadings with reduced radial leakage of neutrons are applied in Czechoslovakia to increase the operational safety and efficiency of WWER-440 type reactors. This method was first used in 1983 at unit 1 of the Bohunice V-1 nuclear power plant. The approximately 15% decrease in the neutron leak allowed the fuel cycle to be extended by 24 effective days. At present, this approach is combined with other ways of increasing the efficiency of operation of WWER-440 type reactors. The software developed enables the fuel loadings to be optimized so that the majority of the fuel is exploited in the reactor for 4 years, and the reactor reload period is extended to 18 months. (Z.M.). 10 figs., 4 tabs., 12 refs.

90
Report of State Surveillance over Nuclear Safety, Czechoslovak Atomic Energy Commission, on a complex assessment of nuclear safety of the V-1 nuclear power plant. *Jaderna Energie (Czechoslovakia);* 37: No. 7, 243-255 (Jul 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

The results of Czechoslovak and foreign experts' assessments of nuclear safety of the V-1 nuclear power plant in Jaslovske Bohunice are summarized. Based on them, it was decided that in its present condition the plant would be operated till the end of 1991. A detailed outline is given of provisions for increasing nuclear safety of this power plant that must be met for the plant to be permitted to continue its operation till the end of 1992. If the reconstruction of the two units is complete, the power plant will be operated till the end of 1995. After this time limit, continued operation of the plant will only be permitted provided that the next stage of reconstruction brings its nuclear safety to the European standard. (Z.M.).

91
Corrosion problems and corrosion resistance of materials for components of PWR nuclear power plants. I: Primary coolant circuit components. Eremias, B. *Jaderna Energie (Czechoslovakia);* 37: No. 7, 256-265 (Jul 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at USD 10.- per typewritten page.

The present insight into corrosion problems of materials for WWER primary circuit components is summarized. These concern zirconium alloys for fuel element casings, steels and alloys for reactor pressure vessels, primary circuit pipes and fittings, and steam generators. Particular attention is paid to corrosion problems of materials for highly stressed components and to the effect of their heat treatment during manufacture. (Z.M.). 10 figs., 7 tabs., 17 refs.

92
A model of crud particle/wall interaction and deposition in a pressurized water reactor primary system. Dinov, K.A. *Nuclear Technology (United States);* 94: No. 3, 281-285 (Jun 1991).

This paper discusses the relative importance of the soluble and particle fractions that take part in the transport of pressurized water reactor corrosion produce in terms of modeling research. A model is proposed that considers the dominant role of the colloidal/particle fraction in the primary coolant system mass and radioactivity transfer. A new hypothesis for a two-stage sticking mechanism is used to quantify the thermal and water chemistry effects on particle/wall interaction. Analytical expressions for the deposition and release coefficients are derived. The results obtained by the MIGA computer code using this model are compared with observations.

93
Properties of the WWER-440 steam generator feedwater circuit derived from temperature and vibration measurements. Matal, O.; Vizek, L.; Beroun, J.; Frelich, P. *Jaderna Energie (Czechoslovakia);* 37: No. 6, 208-213 (Jun 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

It is demonstrated that the data of five operational temperature sensors mounted (following the standard design) to the outer surfaces of pressure vessels of each of the WWER-440 steam generators, do not provide sufficient information to derive the properties of the feedwater circuit or the actual local temperature changes of the steam generator structure. However, if additional temperature and vibration sensors are attached at specific nodes and incorporated into the operation diagnostic system, it is possible to derive the technical condition and properties of the feedwater circuit including the electric feeding pumps, emergency feeding pumps, gate and control valves, flow controllers, heatup systems and thermal insulation. In addition, the data can be employed when designing amendments and modifications of operation regulations and guidelines, particularly with the aim to reduce or eliminate rapid temperature changes. Time dependences of local temperatures are treated to demonstrate some properties of the feedwater circuit of a particular WWER-440 unit during water makeup, preheating of the primary circuit, during a low unit power, start-up of the first feeding pump, etc. (author). 7 figs., 9 refs.

94
Westinghouse's APWR 1000. Stach, V. *Jaderna Energie (Czechoslovakia);* 37: No. 6, 221-224 (Jun 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

The project is described of the advanced pressurized water reactor (APWR) developed, based on cooperation of the American firm Westinghouse with the Mitsubishi Heavy Industries (Japan). The APWR 1000 reactor is being prepared for the American and European markets. Simplification of the systems and reduced consumption of materials, building structures and components, along with a shortening of the time of construction (48 months in the USA), bring about an approximately 20% reduction in the capital costs against the present design of the same power. The operating costs shall also be lower in all the three components; out of this, the total fuel cost reduction shall be about 10%. The design lifetime of a unit is 60 years. (Z.M.).

95
A comprehensive in-pile test of PWR fuel bundle. Kang Rixin; Zhang Shucheng; Chen Dianshan. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 227-233 (Feb 1991). (CONF-900625--).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

An in-pile test of PWR fuel bundle has been conducted in HWRR at IAE of China. This paper describes the structure of the test bundle (3x3-2), fabrication process and quality control of the fuel rod, irradiation conditions and the main Post Irradiation Examination (PIE) results. The test fuel bundle was irradiated under the PWR operation and water chemistry conditions with an average linear power of 381 W/cm and reached an average burnup of 25010 MWd/tU of the fuel bundle. After the test, destructive and non-destructive examination of the fuel rods was conducted at hot laboratories. The fission gas release was 10.4-23%. The ridge height of cladding was 3 to 8 μm. The hydrogen content of the cladding was 80 to 140 ppm. The fuel stack height was increased by 2.9 to 3.3 mm. The relative irradiation growth was about 0.11 to 0.17% of the fuel rod length. During the irradiation test, no fuel rod failure or other abnormal phenomena had been found by the on-line fuel failure monitoring system of the test loop and water sampling analysis. The structure of the test fuel assembly was left undamaged without twist and detectable deformation. (orig.).

96
Fuel rod enrichment determination by gamma scanning. Mainy, P.; Morin, C. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 261-265 (Feb 1991). (CONF-900625--).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

In more than ten years, FBFC has checked the enrichment homogeneity of about five million PWR fuel rods, building up a total experience of nearly thirty-five facility-years. After summarizing measurement principles, the paper outlines the current status of this experience: process efficiency, reliability and drawbacks are described, together with other - simultaneously - inspectable fuel characteristics. The paper also reviews on-going improvements and plans for adapting the examination facility to accomodate changes in the characteristics of future fuel elements. (orig.).

97
Structural analysis of Japanese PWR steel containment vessel under internal pressure loading. Isozaki, Toshikuni; Soda, Kunihisa; Miyazono, Shohachiro. *Nuclear Engineering and Design (Netherlands);* 126: No. 3, 387-393 (May 1991). (CONF-880616–).

From 4. workshop on integrity of containments for nuclear power plants; Arlington, VA (United States) (14-17 Jun 1988).

Elastic-plastic structural analyses of a typical Japanese PWR steel containment vessel under static or dynamic internal pressure loading representing conditions of a typical severe accident were performed using the finite-element analysis code 'ADINA'. In the analysis, static pressure was applied up to 1 MPa, simulating the conditions of a severe accident. Dynamic pressure loadings were assumed, such as a triangular pressure pulse with 10 ms duration and 1 or 2 MPa peak pressure, representing dynamic conditions of hydrogen burn or steam explosion in a containment. It was found in the present analysis that the containment behaves elastically as a whole up to 0.8 MPa in the statically applied loading. (orig.).

98
The results of postirradiation examinations of VVER-1000 and VVER-440 fuel rods. Dubrovin, K.P.; Ivanov, E.G.; Strijov, P.N.; Yakovlev, V.V. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 306-311 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

The paper presents the results of postirradiation examination of fuel rods having different fuel-cladding gaps, pellet densities, pellet inner diameters and so on. The fuel rods were irradiated in the material science reactor (MR) of the Kurchatov Institute of Atomic Energy and at 4 unit of the Novo-Voronezh nuclear powerplant. Some data on fission gas release and rod geometry and compared with computer code predictions. (orig.).

99
Generralized nyquist criterion for the stability of xenon oscillation. Choi, You Cho; Park, Goon Cherl; Chung, Chang Hyun; Park, Chong Kyun. *Journal of the Korean Nuclear Society (Wonjaryok Hakhoeji) (Korea, Republic of);* 22: No. 4, 371-379 (Dec 1990).

The Xenon spatial oscillation may give rise to operational difficulties in a nuclear power plant. In this study,in order to investigate the Xenon instability for a PWR,the frequency-do-main technique is adopted by using Generalized Nyquist Criterion,which is more general and suitable for the multi-input system. Also linearized modal fluxes are obtained by a modal expansion. This modal has been implemented to test the axial Xenon stability of YGN-1 unit against the changes in plant operating parameters;power level, control rod posion, and core average burnup. The results show that the increase of power level and the deeper insection of control rod have the destabilizing effect, and that the burnup progress makes the core less stable. Also the results show that the overestimation due to modal interaction was found not to be significant.(Author).

100
Improvement in fuel utilization in pressurized heavy water reactors due to increased heavy water purity. Balakrishnan, M.R. *Nuclear Technology (United States);* 94: No. 3, 416-420 (Jun 1991).

This paper reports that in a pressurized heavy water reactor (PHWR), the reactivity of the reactor and, consequently, the discharge burnup of the fuel depend on the isotopic purity of the heavy water used in the reactor. The optimal purity of heavy water used in PHWRs, in turn, depends on the cost of fabricated uranium fuel and on the incremental cost incurred in improving the heavy water purity. The physics and economics aspects of the desirability of increasing the heavy water purity in PHWRs in India were first examined in 1978. With the cost data available at that time, it was found that improving the heavy water purity from 99.80% to 99.95% was economically attractive. The same problem is reinvestigated with current cost data. Even now, there is sufficient incentive to improve the isotopic purity of heavy water used in PHWRs. Admittedly, the economic advantage that can be derived depends on the cost of the fabricated fuel. Nevertheless, irrespective of the economics, there is also a fairly substantial saving in natural uranium. That the increase in the heavy water purity is to be maintained only in the low-pressure moderator system, and not in the high-pressure coolant system, makes the option of achieving higher fuel burnup with higher heavy water purity feasible.

101
Arrangement for centering components of a nuclear reactor. Weber, R. World Intellectual Property Organization Patent 90/03647/A1/. 5 Apr 1990. vp. WIPO, 34, chemin des Colombettes, CH-1211 Geneva (CH).

A nuclear reactor contains fuel elements the head of which can be fastened by means of a framework insert. The fuel elements are kept tightly packed in the container and fastened in this position by means of a centering arrangement with centering pins. To facilitate testing and interchange of the centering pins and to enhance safety, the centering pins are arranged at the head of the fuel elements and openings into which the centering pins can be inserted are provided on the framework insert. (author).

102
Obturation and holding-back device for a leaktight closing plug of a steam generator tube. Lenoble, R.; Billoue, J.P. (to Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Courbevoie (France)). France Patent 2648944/A/. 27 Dec 1990. Filed date 23 Jun 1989. 35p. (In French). Available from Institut National de la Propriete Industrielle, Paris (France).

The device comprises a threaded rod which can screwing in a bore tapped in the core and solidary at one end at a blocking element. This blocking element as an external diameter superior at the diameter of the rod and is engaged in the extremity of the plug opposite at the closing end with means for blocking in rotation or/and in translation the blocking device.

Graphite Moderated

103
(CONF-910808–5)
Use of miniature and standard

specimens to evaluate effects of irradiation temperature on pressure vessel steels. Haggag, F.M.; Nanstad, R.K.; Byrne, S.T. (Oak Ridge National Lab., TN (United States)). [1991]. Contract AC05-84OR21400. 7p. OSTI; NTIS; INIS; GPO Dep. Order Number DE91018849.

From 5. international symposium on environmental degradation on materials in nuclear power systems - water reactors; Monterey, CA (United States) (25-29 Aug 1991).

The effects of neutron irradiation on the steel reactor vessel for the modular high-temperature gas-cooled reactor (MHTGR) are being investigated, primarily because the operating temperatures are low [121 to 210°C (250–410°F)] compared to those for commercial light-water reactors (LWRs) [~288°C (550°F)]. The need for design data on the reference temperature shift necessitated the irradiation at different temperatures of A 533 grade B class 1 plate. A 508 class 3 forging, and welds used for the vessel shell, vessel closure head, the vessel flange. This paper presents results from the first four irradiation capsules of this program. The four capsules were irradiated in the University of Buffalo Reactor to an effective fast fluence of 1 $\times 10^{18}$ neutron/cm^2 [0.68 $\times$ 10^{18} neutron/cm^2 (>1 MeV)] at temperatures of 288, 204, 163, and 121°C (550, 400, 325, and 250°F), respectively. The yield and ultimate strengths of both steel plate materials of the MHTGR Program increased with decreasing irradiation temperature. Similarly, the 41-J Charpy V-notch (CVN) transition temperature shift increased with decreasing irradiation temperature (in agreement with the increase in yield strength). The miniature tensile and automated ball indentation (ABI) test results (yield strength and flow properties) were in good agreement with those from standard tensile specimens. The miniature tensile and ABI test results were also used in a model that utilizes the changes in yield strength to estimate the CVN ductile-to-brittle transition temperature shift due to irradiation. The model predictions were compared with CVN test results obtained here and in earlier work. 5 refs., 11 figs., 6 tabs.

104
(IWGGCR–26)
Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Gas-Cooled Reactors). May 1991. 48p. (CONF-9011211–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345.

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

This report contains the minutes of the meeting, the papers presented as overview of the national programmes in the field of gas-cooled reactors and the main results from discussions on the different items of the agenda. The meeting was attended by 20 members and/or alternates from 9 countries and 2 international organizations. 8 papers were presented. A separate abstract was prepared for each of these papers. Refs, figs and tabs.

105
(IWGGCR–26, pp. 25-27)
The situation regarding HTR development in 1990 in the Federal Republic of Germany. Balthesen, E. (Forschungszentrum Juelich GmbH (Germany, F.R.)). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

The HTR-development programme is strongly affected by factors of the overall situation of energy supply and nuclear technology. It is reported that it was impossible in the last years to achieve a greater involvement of operators on a private basis in an implementation of HTR-technology which is not yet commercialized. The author also reports that the AVR test reactor was shut down at the end of 1988 after 21 years of successful operation, and that after many negotiations it was decided in October 1989 to close down the demonstration power plant THTR in Hamm-Uentrop. The further development, i.e. planning work, assessment of the safety and research and development, was concentrated on the plant concept HTR-500, HTR-module and gas-cooled heating reactors, the status of work on which is described.

106
(IWGGCR–26, pp. 27-33)
Status of the HTGR development program in Japan. Saito, S. (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

According to the revision of the Long-Term Program for Development and Utilization of Nuclear Energy issued by the Japanese Atomic Energy Commission, High Temperature Engineering Test Reactor (HTTR), which is the first HTGR in Japan, will be constructed by the Japan Atomic Energy Research Institute (JAERI) in order to establish and upgrade the technology basis for an HTGR, serving at the same time as a potential tool for new and innovative basic research. The budget for the construction of the HTTR was approved by the Government and JAERI is now proceeding with the construction design of the HTTR, focussing the first criticality in the end of FY 1995. In order to establish and upgrade HTGR technology basis systematically and efficiently, and also to carry out innovative basic research on high temperature technologies, Japan will perform necessary R and D mainly at JAERI, which is a leading organization of the R and D. In addition, in order to promote the R and D on HTGRs more efficiently, Japan will promote the existing international cooperation with the research organizations in foreign countries. (author). 5 figs, 3 tabs.

107
(IWGGCR–26, pp. 33-37)
HTGR development in the United States of America. Fox, J.E. (USDOE, Washington, DC (USA)). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

The status of high temperature gas-cooled reactors (HTGR) development in the United States of America is described, including the organizational structure for the development support, HTGR development programme, and plans for future activities in the field.

108
(IWGGCR–26, pp. 38-43)
State of development of gas cooled reactors in the Union of Soviet Socialist Republics. Grebennik, V.N.; Mosevitskij, I.S.; Sukharev, Yu. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (USSR). Inst. Atomnoj Ehnergii). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

In the context of the programme for the development of gas-cooled reactors in the USSR it is reported that pilot plants with VGR-50 MW(el) and VG-400 MW(el) have been developed up to the stage of engineering design and that now the efforts are concentrated on the project of pilot-commercial reactor plant VGM (PCRP VGM) of a modular type with unit thermal power of 200-250 MW. The installation is designed to solve the main scientific and engineering problems of construction of high-temperature gas-cooled reactors, to test equipment components, and to show advantages of the given type of installations having the enhanced safety and capability to generate high-potential heat. The status of work on the PCRP VGM project is described. 3 refs, 1 fig., 1 tab.

109
(IWGGCR–26, pp. 43-45)
Status of the French GCR programmes. Bastien, D. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

It is reported that France has had no research and development programmes for high temperature gas-cooled reactors (HTGR) since 1979, when the decision was taken to end these studies for budgetary reasons. However, recognizing the value of HTGR technology, the French specialists have been continually monitoring developments in this field throughout the world and obtaining necessary scientific and technical experience implementing the HTGR decommissioning programme, the schedule and technological aspects of which are described.

110
(IWGGCR–26, pp. 45-46)
Gas cooled reactor activities in the Commission of the European Communities. Vivante, C. (Commission of the European Communities, Brussels (Belgium)). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

Since the last meeting of the IWGGCR the situation has not changed as far as the CEC activities on gas cooled reactors are concerned. The Commission has no R and D programme in this field but is following and keeping itself informed regarding the developments of this line of reactors. However the studies on the market potential of HTRs as heat and power source in the EEC Countries have continued. A Spanish study is now available, while AEA Technology is doing the same exercise for the UK. It is expected that this latter study will be completed by the end of the year. The AEA study will be the last on the subject. All the studies on this subject which have been carried out in FRG, France, Italy, Spain and the UK will be assembled into a single report which should be available by the end of 1991.

111
(IWGGCR–26, pp. 46)
Status of HTGR activities in Poland. Obryk, E. (Institute of Nuclear Physics, Cracow (Poland)). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

HTGR Technology has been considered in Poland mainly as a future nuclear source of process heat. Research efforts, therefore have been concentrated on high temperature alloys for heat exchanging components. The essential results obtained recently in this field concern structural changes caused by creep and creep-low cycle fatigue interactions and changes due to "hydrogen shocks". It has been established that identical thermal cycling of Incoloy 800 specimens in helium and hydrogen environment generates very different structural changes in both cases. During the last year, a pre-feasibility study of MHTGR applications for co-generation in Poland has been carried out.

112
(IWGGCR–26, pp. 47-48)
HTR development in Switzerland. Brogli, R. (Paul Scherrer Inst. (PSI), Villigen (Switzerland)). May 1991. 48p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603345. (CONF-9011211–).

From International Atomic Energy Agency (IAEA) specialists meeting on behaviour of gas cooled reactor fuel under accident conditions; Oak Ridge, TN (United States) (5-7 Nov 1990).

In Ninth meeting of the International Working Group on Gas Cooled Reactors, Oak Ridge, USA, 8-9 November 1990.

The new activities in the field of high temperature gas-cooled reactors development in Switzerland, taking into account the general situation for nuclear power in the country, are not specifically tuned to a direct project, but are related to the improvement of the HTGR's safety in general, which will be carried out over the next 2-3 years are reported. The nature of this research and development is dedicated to more basic research where a return of industrial investment cannot be immediately foreseen. The main subjects of this basic research are outlined. Participation of Switzerland in the IAEA coordinated research programme "Validation of Safety-Related Physics Calculations for Low-Enriched HTGRs" is also described.

113
(Juel-2421)
Design propositions for top reflectors in high temperature gas cooled reactors of medium size, based on the HTR-500. Otten, J.C. (Forschungszentrum Juelich GmbH (Germany). Inst. fuer Sicherheitsforschung und Reaktortechnik; Technische Hochschule Aachen (Germany)). Jan 1991. 80p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92721507.

Several concepts for top reflectors in medium size high temperature gas cooled reactors (HTGCR's) are investigated. The HTR-500 serves as an example for reactors of this size. The main concern of the investigation is structural strength at high temperatures. The conventional design contains metallic structural parts which are not stable at very high temperatures and might fail during hypothetical accidents. The failing of the support structure would cause the top reflector to fall down on top of the pebble bed core. Closer to the core, it would be more effective, thus it would lead to an increase in reactivity. The thesis contains the results of calculations for two different concepts, one of them being an arch construction, the other one a beam construction. The feasibility of both concepts is evaluated regarding their respective mechanical properties. The arch-based construction shows advantages in its safety and it is easier to build under the given conditions. (orig.).

114
The review of the long term safety of Magnox power stations by the Nuclear Installations Inspectorate. Seddon, J.W.; Pape, E.M.; Goodison, D.M. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 1-10 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273-).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

The history and scope of, and the requirements for, the Magnox long term safety reviews are described and the application of modern standards together with the approach taken to the backfitting of modifications discussed. Some of the lessons learnt from the present Long Term Safety Review Programme are explained and a brief summary of the international position is given. (author).

115
Assurance of safety at CEGB Magnox stations. Parkman, P.R.; Clarke, A.W. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 11-19 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273-).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

The Central Electricity Generating Board's (CEGB) Magnox stations have been subjected to a process of continual safety reviews throughout their lives. Periodic inspections of nuclear safety related components have been carried out to an agreed schedule, numerous specific aspects of plant safety have been rigorously examined and plant improvements which have enhanced specific safety features have been engineered. More recently, Long Term Safety Reviews have been undertaken to complement these regular reviews and to provide confidence that the engineered safety standards remain sufficiently close to those applied to new reactors as to permit continued operation in the longer term. The purpose of this paper is to describe the ongoing safety review process as it applies to Magnox stations. It will describe the depth of technical support underpinning the process and will give examples of specific improvements carried out. The paper will go on to describe the scope of the Long Term Safety Reviews and the arrangements set up to manage them within CEGB so that they may be seen within the proper perspective of the overall assurance of safety. The CEGB's overall conclusions will be presented. (author).

116
Calder Hall and Chapelcross Magnox reactors - a review of safety after 30 years. MacDougall, D.E.; Evans, A.D.; Ayres, G.P. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 21-28 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273-).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

The eight reactors at Calder Hall and Chapelcross are now all around 30 years old and as such are the lead magnox reactors in terms of reactor lifetimes. They were built with large factors of safety and have proved to be simple, extremely reliable and easy to operate. An initial safety review was completed to the Nuclear Installations Inspectorate's satisfaction in 1983. A further review is now nearing completion. The paper describes some of the more interesting aspects of the second review which supports the intention of British Nuclear Fuels plc of an operating life of at least 40 years. (author).

117
Prediction of possible reinforcement corrosion during remaining life, in concrete structures at nuclear installations. Pocock, D.C. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 59-68 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273-).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

Current suggestions are that the complete dismantling and removal of reactors and associated buildings and a return of the sites to green field conditions (Stage 3 decommissioning) may be delayed for periods ranging from several decades to about 100 years after the plant has ceased electricity generation. The time delay is anticipated to lead to reduction in radioactivity levels and thereby to make the dismantling process easier. Throughout their remaining operating life and this delay period the structures are all expected to perform their intended functions efficiently. As existing power stations have been designed for a comparatively shorter working life there is a requirement to assess and predict the long-term performance of these structures. This programme aims to determine the present state and then predict the long-term performance of structures comprising nuclear power plants. (author).

118

One station's experience of the long term safety review. Whitehead, M.J. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 69-78 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273–).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

This paper presents a case history of the Long Term Safety Review at Oldbury Magnox Nuclear Power Station. The scope of the Review is traced from initial guidelines to completion. The paper describes how the comparison with modern standards was tackled. The management of the Review is described, making reference to assistance provided by specialists off-site. Examples of safety improvements arising from the Review are given, with comment on evaluation and cost benefits. The paper concludes that the plant compares well with modern standards despite the apparent advances in safety philosophy since design. (author).

119

Caring for the graphite cores. Wickham, A.J. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 79-87 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273–).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

The extensive Central Electricity Generating Board experience in monitoring the chemical, physical and mechanical properties of Pile Grade 'A' graphite irradiated in Magnox reactors is discussed in terms of the related fault studies and the assessment of mechanical integrity. (author).

120

The application of modern safety standards and methods to Magnox reactors. Munro, H.G.; Board, S.J.; MacArthur, A.A.J. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 49-57 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273–).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

The original design philosophy for the Magnox stations was based on a deterministic approach. The requirements of modern safety standards and the type of results obtained when comparing Magnox plant against them are discussed. Particular attention is given to an example of a probabilistic safety assessment. Stimulated by the Chernobyl accident, the Central Electricity Generating Board is now extending its studies to extreme conditions, to ensure that there are no "cliff edges" just beyond the design basis, and to develop a basis for management of such very low probability events. This work is described and examples given of results obtained. (author).

121

Assessment of the structural integrity of the biological shield at a typical Magnox nuclear power station. Tyrer, M.J. (Nuclear Energy Committee, London (United Kingdom); Institution of Nuclear Engineers, London (United Kingdom); British Nuclear Energy Society, London (United Kingdom)). pp. 101-109 of The review of safety at Magnox nuclear installations. London (UK); Institution of Mechanical Engineers ([1991]). 109p. (CONF-8903273–).

From Seminar on the review of safety at Magnox nuclear installations; London (United Kingdom) (9 Mar 1989).

This paper describes the methods and results of an assessment carried out to evaluate the structural integrity of the biological shield of a typical Magnox type reactor in the very remote event of a catastrophic failure of the steel pressure vessel surrounding the core. Simple static calculations have been carried out to assess the behaviour of the structure in this instance. The behaviour of the pilecap was shown to dominate the failure mode of the biological shield. Even in this very severe accident major bioshield damage is unlikely. Pile cap lifting could only occur in the very worst case. (author).

122

Examination and long-term assessment of nuclear power plant structures. Dawson, P.; Pocock, D. pp. 136-140 of Decommissioning and demolition 1990. Whyte, I.L. (ed.). London (UK); Thomas Telford Ltd. (1990). 254p. (CONF-9004114–).

From 2. international conference on decommissioning offshore, onshore demolition and nuclear works; Manchester (United Kingdom) (24-26 Apr 1990).

Organised by Institution of Civil Engineers, North Western Association; University of Manchester Institute of Science and Technology; University of Manchester.

This paper is primarily concerned with the concrete structures which enclose the reactors of nuclear power plants. These structures are of two types; reinforced concrete biological shields which surround the early Magnox reactors housed in steel pressure vessels, and prestressed concrete pressure vessels (PCPVs) which fulfil the dual role of biological shield and pressure vessel for the later Magnox and the Advanced Gas-Cooled Reactors (AGRs). The author's company has accumulated substantial experience, over thirty years, in the design, surveillance and assessment of these structures. This experience is now directed towards providing assurance of their on-going serviceability during their operational life and their essential functions during the envisaged decommissioning stages for the nuclear plant. This paper summarises the experience obtained by the authors in the examination of nuclear structures and the methods which have been developed to predict their future performance. This recognises that the environment for these structures may change once the reactors they are enclosing are taken out of service. (author).

123

Planning and management of stage 1 dismantling of B16 pile chimney, Sellafield. Shiell, A.E.; Mathews, R.F. pp. 147-155 of Decommissioning and demolition 1990. Whyte, I.L. (ed.). London (UK); Thomas Telford Ltd. (1990). 254p. (CONF-9004114–).

From 2. international conference on decommissioning offshore, onshore demolition and nuclear works; Manchester (United Kingdom) (24-26 Apr 1990).

Organised by Institution of Civil Engineers, North Western Association; University of Manchester Institute of Science and Technology; University of Manchester.

The paper describes the planning and execution of the decommissioning and dismantling of the top sections of the two Pile Chimneys at Sellafield. It describes the complex structure of the Chimneys, their history and original function, and summarises the investigations into the condition of the chimneys, which led to the decision to decommission and the resultant study into the alternative methods to achieve this. Specific mention is made of the approach to safety, both radiological and industrial. This includes preparation of safety cases, environmental protection methods, detailed method statements for the work and safety equipment used. It also explains the management structure and contractual arrangements used to control both safety and cost. The paper then describes the six specific phases of the work, and how each was carried out. The phases of the work are: (1) preparatory work of removing redundant equipment and provision of access to the Chimney top; (2) removal of glass fibre insulation from the cavity above the filters at the Chimney top; (3) protection measures to surrounding plant and buildings; (4) erection of scaffolding around the head of the Chimney and installation of working platforms within the Chimney flue; (5) Removal of the aluminium lining from above the filters and the Chimney cap plates; (6) removal of brickwork and structural steel frame of the Upper and Concentrator Sections. (Author).

124

Decommissioning of the Niederaichbach power plant. Birkhold, U. pp. 189-195 of Decommissioning and demolition 1990. Whyte, I.L. (ed.). London (UK); Thomas Telford Ltd. (1990). 254p. (CONF-9004114–).

From 2. international conference on decommissioning offshore, onshore demolition and nuclear works; Manchester (United Kingdom) (24-26 Apr 1990).

Organised by Institution of Civil Engineers, North Western Association; University of Manchester Institute of Science and Technology; University of Manchester.

Decommissioning of nuclear facilities involves a number of technical and economic problems, such as safety and approval problems, radiation shielding measures, handling and dismantling of large activated components, decontamination methods and release of material. It was decided to study the problems with the decommissioning of the nuclear power plant in Niederaichbach (KKN) to gather experience in the Federal Republic of Germany. KKN is a good example for demonstration as it was shut down in 1974 and has already been transferred to the condition "safe enclosure" and due to its relatively small activity inventory (about 8×10^{13} Bq). Demolition is planned to be in two phases. Phase 1 includes the preparations up to the start of work on site, such as project and planning work, approval procedure as well as construction and fabrication of special tools and manipulators. Phase 1 was finished in June 1987. Phase 2 covers the dismantling work on site and will be completed by return to a "green field" site. Phase 2 started in July 1987 and will be finished in the middle of 1994. (Author).

125

Perspective of the modular high-temperature gas-cooled reactor. Simon, W.A.; Wistrom, J.D.; Holm, B.H. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 299-311 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

The Modular High Temperature Gas-Cooled Reactor (MHTGR) which is being developed in the U.S. has progressed to the preliminary design stage. The design is based on proven technology which has been developed and applied in light water reactors as well as operating gas-cooled reactors, but demonstration of performance is needed prior to extensive commercial deployment. This paper summarizes the principal design and safety features of the MHTGR and project development activities directed towards commercialization. (orig.).

126

Preparations for decommissioning the Windscale piles. Clarke, W.H.; Boorman, T. pp. 235-243 of Decommissioning and demolition 1990. Whyte, I.L. (ed.). London (UK); Thomas Telford Ltd. (1990). 254p. (CONF-9004114–).

From 2. international conference on decommissioning offshore, onshore demolition and nuclear works; Manchester (United Kingdom) (24-26 Apr 1990).

Organised by Institution of Civil Engineers, North Western Association; University of Manchester Institute of Science and Technology; University of Manchester.

The two Windscale piles, whose distinctive chimneys have dominated the skyline of the Sellafield Site since their completion in 1950, were built to produce military material for Britain's nuclear deterrent. This they did successfully until October 1957 when Pile No 1 caught fire during a routine release of the energy which builds up in graphite moderators operated at low temperatures. The fire was put out and the decision taken to cease operations with both of the Piles. Since that date they have been taken to an effective stage 1 decommissioning condition and maintained in a safe state. (Author).

127

Some applications of gamma absorptiometry and spectrometry for the control of nuclear materials. Guery, M. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 266-273 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

In nuclear fuels, and neutrons absorbers used in control rods, the thermal power generated is locally dependent on the concentration of the fissile or absorbing nucleus. In order to control the homogeneity of such materials, non-destructive methods using either gamma absorptiometry or gamma spectrometry were developed; some applications of these methods are presented in this paper. The fuel of the High Temperature Reactor (HTR) is frequently composed of UO_2 and ThO_2 spherical particles dispersed in a carbon matrix; the axial distribution of the particles along the fuel rods can be controlled in two ways: with gamma absorptiometry the heavy elements atoms (U + Th) can be detected but without discrimination between U and Th; with gamma spectrometry, separate distributions of uranium and thorium, deduced from the intensity of characteristic gamma rays are obtained. In nuclear power plants (PWR, FBR) the control rods are made usually with boron carbide (B_4C) pellets. By mean of gamma absorptiometry scanning the density distribution along the axis and the radius of the pellets are obtained. The originality of the method consists in the use of a self-calibration process, then the knowledge of the mass absorption coefficient is not required to perform the examinations. A computerized apparatus has been developed for these controls. (orig.).

128
Computation of heat transfer through a horizontally arranged HTR-fiber insulation. Achenbach, E. *Nuclear Engineering and Design (Netherlands);* 126: No. 3, 379-386 (May 1991). (CONF-891004--).

From 4. international topical meeting on nuclear reactor thermal-hydraulics (NURETH-4); Karlsruhe (Germany) (10-13 Oct 1989).

The heat transfer through a horizontally arranged ceramic fiber insulation operating at high pressure and temperature is studied theoretically. The contributions of heat conduction, natural convection and thermal radiation are considered. With an increasing value of the product of the Rayleigh and Darcy numbers, (Ra, Da), the natural convection is amplified, thus diminishing the efficiency of the thermal insulation. For high temperatures the thermal radiation is no longer negligible. The results of the computer code agree well with experiments available up to a temperature of 670 K and up to a pressure of 40 bar in helium or air. Finally, the heat transfer of a fiber insulation operating under high temperature reactor conditions is predicted, demonstrating that natural convection is nearly suppressed when an insulant bulk density of ρ_b=165 kg/m^3 is provided. (orig.).

129
Nuclear reactor-based power source. Tan, W.P.S. UK Patent 2234849/A/. 13 Feb 1991. 24p. Available from The Patent Office, Sales Branch, Unit 6, Nine Mile Point, Cwmfelinfach, Cross Keys, Newport, NP1 7HZ.

A nuclear reactor-based power source operates on a closed Brayton Cycle using helium or helium/xenon coolant gas with a particle fuel reactor. The reactor is selectively operable to provide different output modes, one of which involves the production of relatively short duration, high energy output bursts. The power source is provided with thermal storage materials having a range of melting points and high latent heats of fusion so that large amounts of waste heat can be absorbed during the burst mode of operation. The reactor has a selectively operable radiator associated with its coolant circuit so that the extent of waste heat dissipation can be varied according to the mode of operation of the reactor. Heat pipes couple waste heat to the radiator. The core comprises circular fuel rods each containing enriched fuel particles dispersed between concentric porous tubes of a frit material. Tube core is surrounded by an annular reflector in which rotatable drums of reflector material are arranged, each drum including a peripheral sliver of neutron absorber for control purposes. Additional control is provided by a central control rod comprising a neutron multiplier section and a neutron absorber section. (author).

Otherwise Moderated or Unmoderated

130
(ECN-RX--91-026)
Sensor failure detection in dynamical systems by Kalman filtering methodology. Ciftcioglu, O.; Turkcan, E. (Netherlands Energy Research Foundation, Petten (Netherlands)). Mar 1991. 9p. (CONF-911014--). OSTI; NTIS (US Sales Only); INIS. Order Number DE92603373.

From International conference on dynamics and control in nuclear power stations; London (United Kingdom) (22-24 Oct 1991).

Design of a sensor failure detection system by Kalman filtering methodology is described. The method models the process systems in state-space form, the information on each state being provided by relevant sensors present in the process system. Since the measured states are usually subject to noise, the estimation of the states optimally is an essential requirement. To this end the detection system comprises Kalman estimation filters, the number of which is equal to the number of states concerned. The estimated state of a particular signal in each filter is compared with the corresponding measured signal and difference beyond a predetermined bound is identified as failure, the sensor being identified/isolated as faulty. (author). 19 refs.; 8 figs.; 1 tab.

131
(IWGFPT--36, pp. 57-66)
Power ramp performance of UO_2 fuel at extended burnup. Hastings, I.J.; Smith, A.D.; Carter, T.J.; Miller, G.C.; Lusk, I.A.; Moeller, R.E.; Rose, D.H. (Atomic Energy of Canada Ltd., Chalk River, ON (Canada). Chalk River Nuclear Labs.). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387--).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

Extending burnup is a practical way to improve the economics of water-reactor operation, via improved fuel utilization and reduced spent-fuel volumes. Atomic Energy of Canada Limited (AECL) is currently focusing on the lightly enriched uranium (SEU) fuel cycle and the CANFLEX advanced fuel bundle. With 1.2% SEU in the CANDU reactor there is up to 35% savings in annual fuel costs. The corresponding core-average burnup is 22 MW.d/kgU, with a maximum element burnup of about 35 MW.d/kgU. Recovered uranium and uranium/plutonium (TANDEM Cycle) from Light-Water Reactors (LWRs) also have economic attraction. Additionally, the capability of a minimum of two hundred and fifty, 100% to 50% full-power, load-following cycles over the life of a fuel bundle is anticipated. Defect-resistance to power ramps at extended burnup is also required. To support AECL's ongoing fuel development program, a power-ramp test was performed on high-burnup, 19-element bundle DG035 from the NPD reactor, since decommissioned. This is one of the 1.4 wt% enriched (U-235 in U) driver bundles that remained in NPD to a maximum final burnup of about 35 MW.d/kgU on the outer elements, corresponding with an in-reactor residence time of 14.5 years. In the U-2 loop power ramp performed in the NRU reactor, the outer elements achieved a maximum linear rating of 32 kW/m, versus a final operating power in NPD of 14 kW/m. The irradiation lasted for 12 days and there were no defects. Post-ramp examination showed up to 10% fission-gas release in outer elements as a result of the ramp, and the formation of intergranular fission-gas bubbles up to 2 μm in diameter. Increased diametral strain was also observed. The test is important to the data based, as existing extended-burnup ramp information is limited. (author). 17 refs, 12 figs, 2 tabs.

132
(IWGFPT–36, pp. 191-195)
Calculation study of the CANDU type fuel performance at extended burnup. Horhoianu, G.; Moscalu, D.R. (Institute for Nuclear Power Reactors, Pitesti (Romania)). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

In the case of CANDU type fuel there is a necessity to improve the fuel bundle design in order to increase its ability to meet high burnup. Further bundle sub-division and slightly enriched uranium is one possibility. Furthermore, grading the element sizes also minimizes the linear power output of the individual fuel elements, while maximizing the power output of the bundle. Also, there are design changes of the pellet that can improve performance at higher burnups. The purpose of this study was to make a comparison between new fuel concept performance and standard CANDU type fuel performance. The results of the thermomechanical behaviour included in this study demonstrate the influence of geometrical and microstructural parameters on the fuel performance. The calculations were performed by means of a probabilistic analysis system developed in INPR. The results of the analysis point out the performances of the new concept and constitute a guideline for future research. (author). 7 refs, 5 figs.

133
(IWGFPT–36, pp. 196-202)
Evolution of the ELESTRES code for applications to extended burnups. Tayal, M.; Ranger, A.; Singhal, N.; Mak, R. (Atomic Energy of Canada Ltd., Montreal, PQ (Canada). CANDU Operations). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

The computer code ELESTRES is frequently used at Atomic Energy of Canada Limited to assess the integrity of CANDU fuel under normal operating conditions. The code also provides initial conditions for evaluating fuel behaviour during high-temperature transients. This paper describes recent improvements in the code in the areas of pellet expansion and of fission gas release. Both of these are very important considerations in ensuring fuel integrity at extended burnups. Firstly, in calculations of pellet expansion, the code now accounts for the effect of thermal stresses on the volume of gas bubbles at the boundaries of UO_2 grains. This has a major influence on the expansion of the pellet during power-ramps. Secondly, comparisons with data showed that the previous fission gas package significantly underpredicted the fission gas release at high burnups. This package has now been improved via modifications to the following modules: distance between neighbouring bubbles on grain boundaries; diffusivity; and thermal conductivity. The predictions of the revised version of the code show reasonable agreement with measurements of ridge strains and of fission gas release. An illustrative example demonstrates that the code can be used to identify a fuel design that would (a) reduce the sheath stresses at circumferential ridges by a factor of 2-10, and (b) keep the gas pressure at very high burnups to below the coolant pressure. (author). 22 refs, 8 figs.

134
(IWGFPT–36, pp. 202-204)
Design of fuel for improved burnup in PHWRs. Anantharaman, K. (Bhabha Atomic Research Centre, Bombay (India). Reactor Engineering Div.). Feb 1991. 216p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605484. (CONF-9006387–).

From Technical committee meeting on fuel performance at high burnup for water reactors; Studsvik (Sweden) (5-8 Jun 1990).

In Fuel performance at high burnup for water reactors.

The trend towards an increase in fuel discharge burnup in recent years has been mainly guided by the desire to shrink the volume of spent fuel inventory and to improve the utilisation of uranium resources. The Pressurized Heavy Water Reactors (PHWR) use natural uranium-di-oxide fuel with least amount of structural materials to achieve better neutron economy. The discharge burnup achieved in PHWRs is about 6700 MWd/Te and this leads to large volume of spent fuel inventory. To overcome the problems due to extended storage of spent fuel and to utilize the plutonium in spent fuel, studies are being carried out to use UO2 - PuO2 fuel along with natural UO2 fuel. This paper describes the work carried out in India to achieve higher fuel burnup. (author). 7 refs, 1 tab.

135
Fabrication of high density UO_2 fuel pellets involving sol-gel microsphere pelletisation and low temperature sintering. Ganguly, C.; Basak, U. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 179-183 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

Powder-free sol-gel microsphere pelletisation and low temperature (1473 K) oxidative sintering processes were used in combination for fabrication of high density ($\geq$ 96% TD) UO_2 fuel pellets for pressurised heavy water reactors. The 'international gelation of uranium' process of BARC was modified for preparation of hydrated gel-microspheres of UO_3 containing 'carbon black' pore former. The gel-microspheres were subjected to controlled air-calcination at 973 K, followed by hydrogen reduction to obtain porous, dust-free and free-flowing UO_2 microspheres suitable for direct pelletisation at 225 MPa. Oxidative sintering of these pellets at 1473 K in CO_2 atmosphere followed by Ar + H_2 treatment led to high density ($\geq$ 96% TD) UO_2 pellets having equiaxed grains of $\leq$ 10 μm and uniformly distributed 'closed' spherical pores in the diameter range of 2-5 μm. Resintering of these pellets at high temperature (1973 K) for 8 hours in Ar + 8% H_2 atmosphere did not show any significant change in pellet dimension or grain size. The UO_2 pellets prepared by sol-gel microsphere pelletisation route had higher thermal conductivity compared to pellets of equivalent density prepared by the 'powder-pellet' route. UO_2 pellets of large grain size (45-55 μm) and high density could be obtained with TiO_2 dopant and high temperature sintering in Ar + H_2 atmosphere. TiO_2 dopant was not effective for low temperature oxidative sintering. (orig.).

136
Core simulations using actual detector readings for a Canada deuterium uranium reactor. Kim, I.S.; Kim, S.Y.;

Kim, B.G. *Nuclear Technology (United States);* 93: No. 2, 138-146 (Feb 1991).

This paper reports that, to obtain better simulation results for a Canada deuterium uranium (CANDU) reactor operation, a new simulation method is developed that uses actual detector readings as a correction factor. Detector readings from a CANDU reactor are used to correct the calculated flux distribution during core calculation iterations. A suitable function is found to describe the relationship between the detector flux and the fluxes of mesh points around the detector. The new simulation method is tested by performing numerical calculations for the Wolsung reactor (a CANDU-600). The results show that the new method predicts the core state more accurately with fewer iterations.

137
Characterization of HWR fuel pellets fabricated using UO_2 powders from different conversion processes. Lee, Y.W.; Yang, M.S. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 217-226 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

Heavy-water-reactor (HWR) fuel manufacturing involves the fabrication of high density natural UO_2 pellets with controlled microstructure. It is necessary to investigate the correlation between powder characteristics, pellet properties and process parameters, especially when the UO_2 powders are from different conversion processes. To establish the UO_2 pelletizing technology better to obtain such tightly quality controlled pellets, it is found necessary, apart from the quality requirements described in various specifications, to determine the particle morphology, the pore size distribution in particular, with regard to the compaction behaviour, which strongly affects the densification during sintering. Pore structures of different UO_2 powder converted via the AUC and ADU processes and the evolution of pore size distribution in the compacted pellets from these different UO_2 powders are analysed and compared by mercury porosimetry and microstructure observations of compacted pellets by scanning electron microscopy. Further, the connection of the evolution of pores in compacted pellets to the attainable densities of sintered pellets is discussed for the optimization of process parameters and as a proposal for quality control requirements and method. (orig.).

138
Quality evaluation and control of end cap welds in PHWR fuel elements by ultrasonic examination. Choi, M.S.; Yang, M.S. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 321-327 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

The current quality control procedure of nuclear fuel end cap weld is mainly dependent on the destructive metallographic examination. A nondestructive examination technique, i.e., ultrasonic examination, has been developed to identify and evaluate weld discontinuities. A few interesting results of the weld quality evaluation by applying the developed ultrasonic examination technique to PHWR fuel welds are presented. In addition, the feasibility of the weld quality control by the ultrasonic examination is discussed. This study shows that the ultrasonic examination is effective and reliable method for detecting abnormal weld contours and weld discontinuities such as microfissure, crack, upset and expulsion, and can be used as a quality control tool for the end cap welding process. (orig.).

139
Technical and economic evaluations of CANDU advanced fuel bundle designs. Suk, H.C.; Hwang, W.; Park, J.H.; Kim, B.G.; Sim, K.S.; Jeong, C.J.; Heo, Y.H.; Jun, J.S. *Journal of the Korean Nuclear Society (Wonjaryok Hakhoeji) (Korea, Republic of);* 22: No. 4, 389-409 (Dec 1990). (In Korean).

As a principal design of advanced CANDU fuel bundle, CANDU-KF39, CANDU-KF40 and CANDU-Kf43 fuel bundles were proposed and evaluated with respect to the operating conditions of the CANDU-6 reactor of Wolsung Unit-1. From the results, the advanced fuel bundles show to be improved economical and technical benefits compared with the current 37-element bundle. Especially, it was appeared that the KF-39 fuel bundle has more benefits of the safety, technical and economical aspects of Wolsung Unit-1 rather than those of the KF-40 and KF-43 fuel bundles.(Author).

Breeding

140
(ANL/CP–72041)
Fabrication of oxide dispersion strengthened ferritic clad fuel pins. Zirker, L.R.; Bottcher, J.H.; Shikakura, S.; Tsai, C.L.; Hamilton, M.L. (Argonne National Lab., IL (United States)). [1991]. Contract W-31109-ENG-38. 11p. (CONF-911001–21). OSTI; NTIS; INIS; GPO Dep. Order Number DE91018641.

From International conference on fast reactor systems and fuel cycles; Kyoto (Japan) (27 Oct - 1 nov 1991).

A resistance butt welding procedure was developed and qualified for joining ferritic fuel pin cladding to end caps. The cladding are INCO MA957 and PNC ODS lots 63DSA and 1DK1, ferritic stainless steels strengthened by oxide dispersion, while the end caps are HT9 a martensitic stainless steel. With adequate parameter control the weld is formed without a residual melt phase and its strength approaches that of the cladding. This welding process required a new design for fuel pin end cap and weld joint. Summaries of the development, characterization, and fabrication processes are given for these fuel pins. 13 refs., 6 figs., 1 tab.

141
(ANL/CP–72603)
Economic prospects of the Integral Fast Reactor (IFR) fuel cycle. Chang, Y.I.; Till, C.E. (Argonne National Lab., IL (United States)). [1991]. Contract W-31109-ENG-38. 6p. (CONF-911001–15). OSTI; NTIS; INIS; GPO Dep. Order Number DE91018622.

From International conference on fast reactor systems and fuel cycles; Kyoto (Japan) (27 Oct - 1 nov 1991).

The IFR fuel cycle based on pyroprocessing involves only few operational steps and the batch-oriented process equipment systems are compact. This results in major cost reductions in all of three areas of reprocessing, fabrication, and waste treatment. This document discusses the economic aspects of this fuel cycle.

142
(ANL/CP–72616)
Operating and test experience of EBR-II. Sackett, J.I. (Argonne National Lab., Idaho Falls, ID (United States)). [1991]. Contract W-31109-ENG-38. 19p. (CONF-911001–20). OSTI; NTIS; INIS; GPO Dep. Order Number DE91018639.

From International conference on fast reactor systems and fuel cycles; Kyoto (Japan) (27 Oct - 1 nov 1991).

EBR-2 has operated for 27 years, the longest for any Liquid Metal Reactor (LMR) power plant. During that time, much has been learned about successful LMR operation and design. The basic lesson is that conversatism in design can pay significant dividends in operating reliability. Furthermore, such conservatism need not mean high cost. The EBR-2 system emphasizes simplicity, minimizing the number of valves in the heat transport system, for example, and simplifying the primary heat-transport-system layout. Another lesson is that emphasizing reliability of the steam generating system at the sodium-water interface (by using duplex tubes in the case of EBR-2) has been well worth the higher initial costs; no problems with leakage have been encountered in EBR-2's operating history. Locating spent fuel storage in the primary tank and providing for decay heat removal by natural connective flow have also been contributors to EBR-2's success. The ability to accommodate loss of forced cooling or loss of heat sink passively has resulted in benefits for simplification, primarily through less reliance on emergency power and in not requiring the secondary sodium or steam systems to be safety grade. Also, the "piped-pool " arrangement minimizes thermal stress to the primary tank and enhances natural convective flow. These benefits have been realized through a history of operation that has seen EBR-2 evolve through four major phases in its test programs, culminating in its present mission as the Integral Fast Rector (IFR) prototype. 20 refs., 8 figs., 1 tab.

143
(ANL/CP–73613)
Reliability of fast reactor mixed-oxide fuel during operational transients. Boltax, A.; Neimark, L.A.; Tsai, Hanchung; Katsuragawa, M.; Shikakura, S. (Argonne National Lab., IL (United States)). Jul 1991. Contract W-31109-ENG-38. 28p. (CONF-911001–18). OSTI; NTIS; INIS; GPO Dep. Order Number DE91018633.

From International conference on fast reactor systems and fuel cycles; Kyoto (Japan) (27 Oct - 1 nov 1991).

Results are presented from the cooperative DOE and PNC Phase 1 and 2 operational transient testing programs conducted in the EBR-2 reactor. The program includes second (D9 and PNC 316 cladding) and third (FSM, AST and ODS cladding) generation mixed-oxide fuel pins. The irradiation tests include duty cycle operation and extended overpower tests. the results demonstrate the capability of second generation fuel pins to survive a wide range of duty cycle and extended overpower events. 15 refs., 9 figs., 4 tabs.

144
(INIS-BR–2813)
On modelling, mathematical analysis and numerical treatment of three-dimensional transient two-phase coolant flow in engineering systems. Bottoni, M.; Sengpiel, W. (Argonne National Lab., IL (USA)). (Associacao Brasileira de Ciencias Mecanicas, Rio de Janeiro, RJ (Brazil)). 1990. 6p. (CONF-9012116–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92605542.

From 3. National Meeting of Thermal Sciences; Itapema (Brazil) (10-12 Dec 1990).

The conservations equations of mass, momentum and enthalpy and the entropy inequality, written for a two-phase flow in the local instantaneous form together with the respective jump conditions at the phase interfaces, are volume-averaged over Eulerian control cells yielding a system of coupled macroscopic governing equations for the separated phases suitable for the three-dimensional, transient numerical simulation of complex engineering systems. The equations for the separated phases are then combined to model the fluid mixture in the frame of a slip model. The state of the art of the numerical treatment of the discretized and linearized equations is presented with reference to solution methods of a resulting Poisson-equation for pressure and enthalpy distributions. The above algorithms have been implemented in the computer programmes BACCHUS-3D/TP and COMMIX-2. The application of these codes is reviewed with reference to the numerical simulation of sodium boiling experiments in bundle geometry and of forced and natural convection simulations in more complex geometrical configurations. (author).

145
(IWGFR–79)
Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990. Girard, J.P. (ed.). (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Oct 1990. 379p. (CONF-9010432–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543.

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

23 officially nominated persons and 8 observers from 7 countries operating fast breeder reactors in the world, Mr. Arkhipov, IAEA Scientific Secretary of IWGFR and Mr. Cambillard, French member of IWGFR attended the specialists meeting. 25 papers were presented in the national status session and in 3 technical sessions devoted to methods, theoretical approach and real steam generator experience. A separate abstract was prepared for each of these papers. Since the last meetings in Dimitrovgrad and Petten it is clear that acoustic/ultrasonic monitoring of in-sodium water leaks is now considered by all countries as a major topic for commercial fast reactor steam generator unit protection. At this time the detection of leakage events is thought to be possible in the leak range from 1 to about 100 g/s in a time period of a few seconds to a few tens of seconds. Future work should aim at a more precise definition of the attainable limits, taking into account the particular requirements of actual plant design. Refs, figs and tabs.

146
(IWGFR–79, pp. 39-46)
Review of the common European R and D programme on acoustic leak detection for steam generators. Voss, J.; Thomas, P.J.; Girard, J.P. (Internationale Atomreaktorbau GmbH (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The SGU protection in EFR will rely on fast acoustic monitoring (amongst other methods). In cases of fast leak propagation acoustic detection should be able to provide an early alarm. However the requested high reliability and low spurious trip rate still represent a severe challenge. During the last years a joint R and D programme has been defined and started. It includes: background nose and simulated leak noise measurements on running reactors; acoustic measurements during sodium water reactions in test rigs; development of both passive and active acoustic methods; improvement of signal processing techniques; concept of EFR prototype leak detection system. The programme, shared among the European R and D partners (AEA, CEA, Interatom), was introduced and a summary of the present status was given. (author). 3 figs.

147
(IWGFR–79, pp. 47-62)
Detection of steam leaks into sodium in fast reactor steam generators by acoustic techniques - An overview of Indian programme. Prabhakar, R.; Vyjayanthi, R.K.; Kale, R.D. (Indira Gandhi Centre for Atomic Research, Kalpakkam (India)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

Realising the potential of acoustic leak detection technique, an experimental programme was initiated a few years back at Indira Gandhi Centre for Atomic Research (IGCAR) to develop this technique. The first phase of this programme consists of experiments to measure background noise characteristics on the steam generator modules of the 40 MW (thermal) Fast Breeder Test Reactor (FBTR) at Kalpakkam and experiments to establish leak noise characteristics with the help of a leak simulation set up. By subjecting the measured data from these experiments to signal analysis techniques, a criterion for acoustic leak detection for FBTR steam generator will be evolved. Second phase of this programme will be devoted to developing an acoustic leak detection system suitable for installation in the 500 MWe Prototype Fast Breeder Reactor (PFBR). This paper discusses the first phase of the experimental programme, results obtained from measurements carried out on FBTR steam generators and results obtained from leak simulation experiments. Acoustic leak detection system being considered for PFBR is also briefly described. 4 refs, 8 figs, 1 tab.

148
(IWGFR–79, pp. 63-64)
National status on acoustic leak detection for fast breeder reactor in Japan. Higuchi, Masahisa. (Japan Atomic Power Co., Tokyo (Japan). FBR Mechanical and Electrical Section). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The research and development programme on acoustic leak detection for fast breeder reactors in Japan, with emphasis on acoustic wave analysis and background noise analysis, is outlined.

149
(IWGFR–79, pp. 65-76)
Status of U.S. evaluations of acoustic detection of in-sodium water leaks. Fletcher, F.L.; Neely, H.H.; Buschman, H.W. (Rockwell International Corp., Canoga Park, CA (USA). Energy Technology Engineering Center). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

An overview of the United States testing program to evaluate acoustic leak detection and location systems on simulated water leaks in functional liquid sodium steam generators is provided. Testing was conducted on the modular hockey stick steam generator during the large leak test program in the LLTR, on the CRBR prototype hockey stick steam generators in SCTI, on a double wall tube steam generator installed in EBR-II and on the helical coil steam generator tested in SCTI. These test programs have demonstrated the acoustic leak detection system potential, however, additional development is required before the system can perform to its effective and required potential. (author). 6 figs.

150
(IWGFR–79, pp. 91-104)
A feasibility study on active ultrasonic techniques for water into sodium leak detection on FBR steam generator units. Girard, J.P.; Garnaud, P.; Journeau, C.; Demarais, R. (CEA, Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-lez-Durance (France). Direction des Reacteurs Nucleaires). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

In the framework of the European Fast Breeder Project one of the aims is to provide the ferritic straight tube steam generator with a fast and reliable leak detection system. The first studies of water sodium leaks, based on the passive listening of noise source, are described. Considerable experience has been acquired of this technique and one of the conclusions is that a high level of reliability may require a sophisticated surveillance algorithm. Further works on the subject should lead to demonstration phase in 1993-1995 on a real and representative steam generator unit in order to have the benefit of a long term run of the surveillance method prior to industrial use in a compulsory safety system. 1 ref., 10 figs.

151
(IWGFR–79, pp. 105-114)
Criteria for assessing the quality of signal processing techniques for acoustic leak detection. Prabhakar, R.; Singh, O.P. (Indira Gandhi Centre for Atomic Research, Kalpakkam (India). Reactor Group). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS.

Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

In this paper the criteria used in the first IAEA coordinated research programme to assess the quality of signal processing techniques for sodium boiling noise detection are highlighted. Signal processing techniques, using new features sensitive to boiling and a new approach for achieving higher reliability of detection, which were developed at Indira Gandhi Centre for Atomic Research are also presented. 10 refs, 3 figs, 2 tabs.

152
(IWGFR–79, pp. 115-130)
Development of acoustic leak detection system in PNC. Tanabe, H.; Kuroha, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The development of an acoustic leak detector is under way at PNC as a detection system that has potential of quick response and high reliability for larger steam generators of future LMFBR plants. The studies have two aspects, i.e., an acoustic wave analysis in various sodium-water reactions and a background noise (BGN) analysis in a sodium-heated 50MWt steam generator (50MWGS). In the former analysis, wave profiles of the sodium-water reaction sound were analyzed and compared with those of inert gas injection sound. The comparison revealed that there were no wave profiles specific to a sodium-water reaction sound. The latter clarified that major acoustic sources in the steam generator were sodium flow and steam generation/flow and that the water leak rate at which a noise level was comparable with that of the background noise was about 0.5 g/sec. in the evaporator of 50MWSG. The estimation of acceleration levels of BGN and leak sounds in other plants reveals that an intermediate leak is detectable in the Monju evaporator with a present acoustic detection system. (author). 2 refs, 9 figs.

153
(IWGFR–79, pp. 131-144)
On-line low and high frequency acoustic leak detection and location for an automated steam generator protection system. Gaubatz, D.C.; Gluekler, E.L.; Fletcher, F.; Claytor, T. (General Electric Co., San Jose, CA (USA). Nuclear Energy Div.). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

Two on-line acoustic leak detection systems were operated and installed on a 76 MW hockey stick steam generator in the Sodium Components Test Installation (SCTI) at the Energy Technology Engineering Center (ETEC) in Southern California. The low frequency system demonstrated the capability to detect and locate leaks, both intentional and unintentional. No false alarms were issued during the two year test program even with adjacent blasting activities, pneumatic drilling, shuttle rocket engine testing nearby, scrams of the SCTI facility, thermal/hydraulic transient testing, and pump/control valve operations. For the high frequency system the capability to detect water into sodium reactions was established utilizing frequencies as high as 300 kHz. The high frequency system appeared to be sensitive to noise generated by maintenance work and system valve operations. Subsequent development work which is incomplete as of this date showed much more promise for the high frequency system. (author). 13 figs.

154
(IWGFR–79, pp. 145-163)
Acoustic surveillance techniques for SGU leak monitoring. McKnight, J.A.; Rowley, R.; Beesley, M.J. (AEA Technology, London (UK)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The paper presents a brief review of the acoustic techniques applicable to the detection of steam generator unit leaks that have been studied in the UK. Before discussion of the acoustic detection methods a reference representation of the required performance as developed in the UK is given. The conclusion is made that preliminary specification for the acoustic leak detection of sodium/water leaks in steam generating units suggests that it will be necessary to detect better than a leak rate of 3 g/s within a few seconds. 10 refs, 12 figs.

155
(IWGFR–79, pp. 165-193)
Experimental studies on acoustic detection of sodium-water steam generator leaks in the USSR. Petrenko, A.A.; Poplavsky, V.M. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk (USSR). Fiziko-Ehnergeticheskij Inst.). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The paper reports that the acoustic leak indicators have been developed in two versions. The first one is based upon using the immersible acoustic hydrophones and the parallel frequency analysis of their signals. The second one uses the waveguide sensors with microprocessor system of noise signals processing. Brief description of both versions is given. The result of these systems tests at the experimental facilities, BN-600 and BOR-60 reactors are also provided. 4 refs, 15 figs.

156
(IWGFR–79, pp. 195-206)
Sound transmission at the sodium/shell interface modelisation and recommendation for SG leak monitoring. Journeau, C.; Hamidi, M.A.; Ramdane, A. (CEA, Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-lez-Durance (France). Direction des Reacteurs Nucleaires). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The purpose of this paper is to examine the fluid structure interaction at the steam generator (SG) shell. It was simplified as a cylindrical "container" filled with water and excited either by a point force or a point acoustic source. A boundary element modelling of the structure and fluid was performed in order to evaluate the induced shell vibrations. Numerical results are provided and recommendations are drawn. (author). 3 refs, 7 figs.

157
(IWGFR–79, pp. 207-220)
Acoustic monitoring of steam generators in service and water-into-sodium experiments. Fletcher, F.; Claytor, T.N.; Gaubatz, D.C. (Rockwell International Corp., Canoga Park, CA (USA). Energy Technology Engineering Center). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

An overview of the United States program to evaluate acoustic leak detection systems on functional steam generators and during conduction of water-into-sodium experiments is presented. Testing was conducted at GE during the performance of water-into-sodium experiments, at EBR-II with simulated water leaks using the EBR-II double wall tube steam generators; at ETEC during the LLTR test program using water injections into sodium, and during the CRBR prototype hockey stick steam generator and the prototype B and W helical coil test programs using simulated water leaks. The acoustic methodology was demonstrated during these test programs, however, additional development will be required before a system can accomplish its potential and required performance. (author). 12 figs.

158
(IWGFR–79, pp. 221-235)
Sound propagation in the steam generator - A theoretical approach. Heckl, M. (Keele Univ. (UK). Dept. of Physics). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

In order to assess the suitability of acoustic tomography in the steam generator, detailed information on its acoustic transmission properties is needed. We have developed a model which allows one to calculate the sound field produced by an incident wave in the steam generator. In our model we consider the steam generator as a medium consisting of a two-dimensional array of infinitely long cylindrical tubes. They are thin-walled, made of metal and are immersed in a liquid. Inside them there is a liquid or a gas. The incident wave is plane and perpendicular to the cylindrical tubes. When a sound wave crosses the tube bundle, each individual tube is exposed to a fluctuating pressure field and scatters sound which, together with the incident wave, influences the pressure at all surrounding tubes. The motion of an individual tube is given by differential equations (Heckl 1989) and the pressure difference between inside and outside. The interaction of a tube wall with the fluid inside and outside is treated by imposing suitable boundary conditions. Since the cylinder array is periodic, it can be considered as consisting of a large number of tube rows with a constant distance between adjacent cylinders within a row and constant spacing of the rows. The sound propagates from row to row, each time getting partly transmitted and partly reflected. A single row is similar to a diffraction grating known from optics. The transmission properties of one row or grating depend on the ratio between spacing and wavelength. If the wavelength is larger than the spacing, then the wave is transmitted only in the original direction. However, for wavelengths smaller than the spacing, the transmitted wave has components travelling in several discrete directions. The response of one row to sound scattered from a neighbouring row is calculated from Kirchhoff's theorem. An iteration scheme has been developed to take the reflection and transmission at several rows into account.

159
(IWGFR–79, pp. 239-271)
Background- and simulated leak-noise measurements on ASB-loop, KNK II- and SNR 300-steam generators. Voss, J.; Arnaoutis, N.; Foerster, K.; Moellerfeld, H. (Internationale Atomreaktorbau GmbH (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

During several leak propagation experiments in the ASB sodium loop noise measurements were performed showing the acoustic behaviour of evoluting leaks in a tube bundle section under sodium. Effects like self evolution, secondary leaks and tube ruptures by overheating occurred during these tests and were reflected in the course of acoustic signals. In one of the KNK II steam generators simulated leak noise was detected against background noise throughout the operating power range. Experimental arrangements and results are described. In SNR 300 all of the SGUs are equipped with waveguides and some with accelerometers for background noise measurements. First measurement under isothermal conditions were performed in the past. A gas injection device for acoustic leak simulation is under construction. The design of the experimental acoustic system and first results are presented. (author). 1 ref., 21 figs, 2 tabs.

160
(IWGFR–79, pp. 311-320)
Acoustic sodium-water reaction detection of the Phenix steam generators. Carminati, M.; Martin, L.; Sauzaret, A. Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The systems for acoustic sodium-water reaction detection and hydrogen detection of the Phenix steam generators as well as systems for monitoring signals analysis and processing are described. It is reported that the results obtained during operation and calibration phases are very encouraging and that industrial equipment showing the same performance are being examined. 6 figs.

161
(IWGFR–79, pp. 321-330)
Water leak detection in steam generator of Super Phenix. Kong, N.; Brunet, M.; Garnaud, P.; Ghaleb, D. (Electricite de France (EDF), 78 - Chatou (France)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

With the intent of detecting water leaks inside steam generators, we developed a third system, called acoustic detector, to complement hydrogen detectors and rupture disks (burst disks). The role of the acoustic system is to enable rapid intervention in the event of a leak growing rapidly which could rupture neighbouring tubes. In such a case, the detectable flow rate of the leak varies from a few tens of g/s to a few hundred g/s. At the Super Phenix, three teams work in parallel in complementary frequency bands: EDF (0-20 kHz), CEA/SPCI (20-100 kHz) and CEA/STA (50-300 kHz). The simulation of water leaks in the steam generator by the argon injections performed to date at 50% of the rated power has shown promising results. An anomaly in the evolution of the background noise at more than 50% loading of one of the two instrumented steam generators would make difficult any extrapolation to full power behaviour. 5 refs, 6 figs, 1 tab.

162
(IWGFR–79, pp. 331-337)
Acoustic leak detector in Monju steam generator. Wachi, E.; Inoue, T. (Power Reactor and Nuclear Fuel Development Corp. (Japan)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

Acoustic leak detectors are equipped with the Monju steam generators for one of the R and D activities, which are the same type of the detectors developed in the PNC 50MW Steam Generator Test Facility. Although they are an additional leak detection system to the regular one in Monju SG, they would also detect the intermediate or large leaks of the SG tube failures. The extrapolation method of a background noise analysis is expected to be verified by Monju SG data. (author). 4 figs.

163
(IWGFR–79, pp. 338-349)
Analysis of acoustic data from the PFR SGU condition monitor. Rowley, R.; Airey, J. (AEA Technology, London (UK)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

This paper gives an outline description of an acoustic monitoring system which has been installed on the SGU of the Prototype Fast Reactor (PFR) at Dounreay with the objective of giving early warning of any change in noise output which could be related to potentially damaging vibrations within the units. Data obtained from this PFR monitoring system is playing an important part in the development of acoustic instrumentation for leak detection although this had not been the primary objective of this particular installation. The PFR has three secondary circuits each containing an evaporator, a superheater and a reheater giving a total of nine SGUs. Although the design of the units is different from that intended for EFR, the measurements provide a valuable source of information on the character and amplitude of acoustic background noise in operational steam generator units. The vibration monitoring system uses the waveguides originally installed during reactor commissioning for leak detection studies. Twelve acoustic waveguides are fitted to the shell of each of the units. The superheaters and reheaters have three waveguides at each of four axial levels, while the evaporators have four waveguides at each of three axial levels. In addition the evaporators have a small number of waveguides attached to the top flange of the unit. Each waveguide is fitted with an accelerometer to record the acoustic signal from the SGU. Tape recordings of the acoustic noise from each unit are made on a regular basis and the tapes analysed on an automated analysis system which has been developed to extract and store in a database about 20 characteristic features from the data. The paper gives examples of the background noise from the SGU. The data demonstrates the use of location techniques to identify prominent acoustic source. 8 figs.

164
(IWGFR–79, pp. 351-362)
Analysis of acoustic data from UK sodium/water reaction test facilities. Rowley, R.; Mcknight, J.A.; Airey, J. (AEA Technology, London (UK)). Oct 1990. 379p. OSTI; NTIS (US Sales

Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

This paper describes acoustic measurements made during a number of sodium/water reaction experiments in the UK. The tests have included water and steam injections through both realistic (fatigue crack) defects and machined orifices and have covered a range of experimental conditions including those appropriate to the inlet and outlet regions of the EFR steam generators. Injection rates were typically in the range 0.1 to 30 g/s. Where possible, gas injections were also included in the test programme for comparison, since it is anticipated that a practical SGU acoustic leak detection system would include a facility for gas injections to allow system calibration, and to confirm transmission properties within the SGU. The test sections were instrumented with accelerometers on waveguides and in some cases included an under-sodium microphone situated about 300mm above the reaction zone. Tape recordings were made during the tests and used for detailed analysis off-line, although an audible output from one of the acoustic channels was used to monitor the progress of the injections and provide information for the rig operators. A comparison of the signal amplitudes measured during the experiments with typical reactor background noise was made and an estimate of the detection sensitivity of an acoustic monitoring system was deduced. 3 refs, 5 figs, 1 tab.

165
(IWGFR–79, pp. 363-369)
Towards an active acoustic detection system on EFR. Blanc, D.; Villani, D.; Ford, J. (Novatome, Lyon (France)). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The possibilities of an ultrasonic leak detection system used for the EFR project are outlined. 1 ref., 2 figs, 2 tabs.

166
(IWGFR–79, pp. 371-379)
The current status of research and development concerning steam generator acoustic leak detection for the demonstration FBR plant. Higuchi, Masahisa. (Japan Atomic Power Co., Tokyo (Japan). FBR Electrical and Mechanical Section). Oct 1990. 379p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605543. (CONF-9010432–).

From IAEA IWGFR specialists' meeting on steam generators: acoustic/ultrasonic detection of in-sodium water leaks; Aix-en-Provence (France) (1-3 Oct 1990).

In Proceedings of the specialists' meeting on acoustic/ultrasonic detection of in sodium water leaks on steam generators, held in Aix-en-Provence, France, 1-3 October 1990.

The Japan Atomic Power Co. (JAPC) started the research and development into Acoustic Leak Detection for the Demonstration FBR (D-FBR) plant in 1989. Acoustic Leak Detection is expected as a water leak detection system in the Steam Generator for the first D-FBR plant. JAPC is presently analyzing data on Acoustic Leak Detection in order to form some basic concepts and basic specifications about leak detection. Both low frequency types and high frequency types are selected as candidates for Acoustic Leak Detection. After a review of both types, either one will be selected for the D-FBT plant. A detailed Research and Development plan on Acoustic Leak Detection, which should be carried out prior to starting the construction of the D-FBR plant, is under review. (author). 3 figs, 2 tabs.

167
U.S. advanced liquid metal reactor. Berglund, R.C.; Steiner, H. (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 313-323 of Nuclear power: Today, tomorrow. Duesseldorf (Germany); VDI-Verl. (1991). 323p. (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

The U.S. Advanced Liquid Metal Reactor (ALMR) design and licensing program, aimed at a long term, competitive breeder reactor, is being carried out under U.S. DOE sponsorship based on GE's PRISM (Power Reactor Innovative Small Module) concept. The design utilizes a modular approach for improved construction schedules, expedited learning series benefits, lower development costs, inclusive of a full scale demonstration unit, and is projected to result in a commercially competitive power plant with outputs from 465 MWe to 1400 MWe. The design employs numerous innovations with particular emphasis on passive safety features, low investment risk and independence from human operator action. The high reliability of its preventive and mitigative features should allow licensing without the need for abnormal evacuation planning. The ALMR offers a unique potential for actinide recycle such that its commercial deployment early in the 21st century could enhance the solution to the U.S. high level nuclear waste management challenge. (orig./HP).

168
On the relative role of processes whose sequence results in crack growth in the cladding of LMFBR fuel pins. Mikhlin, E.Ya. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 183: No. 3, 180-185 (Aug 1991).

Processes are discussed the joint effect of which results in crack development in austenitic steel-clad oxide fuel pins. Such processes include generation of Te which is considered as the main embrittling agent, its transport and accumulation at the cladding inner surface, where together with Cs it forms a liquid surface-acting medium, and finally, development of intergranular cracks in the cladding caused by the contact with this medium. As the process of crack growth in itself proceeds faster than accumulation of liquid surfactants at the cladding, the cracks will be able to reach the critical length only after the necessary amount of Te has been accumulated. Its accumulation is determined and therefore, controlled by the process of Te transport in the fuel grains. It is shown that the main contribution to the accumulation of Te at the cladding surface is provided by the hottest internal zones of the fuel pellet.

On the basis of the analysis given, means are discussed, for inhibiting or blocking the crack growth. (orig.).

169
Autocatalysis in fast reactor disassembly. Walker, S.P. *Nuclear Engineer (Institution of Nuclear Engineers) (United Kingdom);* 30: No. 3, 149-154 (Jun 1991).

By increasing the effective volatility of the fuel, fission products can influence the course of disassembly accidents. Generally such an influence would be expected to be beneficial, with accident termination after lower energy release. In this Paper a case in which sodium and fission gases are both assumed to have been lost from the central region of the core earlier in the accident is analysed. Such losses, earlier examined separately and found to be benign, are together shown to be able to cause reactivity addition through inwards movement of fuel. Significantly greater energy release is predicted as a result. (Author).

170
Automatic refabrication and quality control of fuel elements for the BOR-60 reactor in the USSR. Steinkopff, H.; Krompass, R.; Skiba, O.V. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 163-170 (Feb 1991). (CONF-900625--).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

The USSR and the former GDR pursued the common objective in a contract about scientific cooperation in order to project, to erect and to take in operation an automated pilot plant for the refabrication of fuel elements for the fast reactor BOR-60. The layout of the plant was determined mainly by the radioactivity and toxicity of the U/Pu oxide fuel and by the application of the procedure of electrodynamic vibration to densify the fuel. For vibrocompacting non-spherical granular U/Pu oxide fuel with a definite spectrum of particle sizes and a sufficient flowability was prepared. Relations between particle size mixture, the density and the axial density distribution of the vibrocompacted fuel column is presented by elected examples. For the quality assurance the fuel elements was controlled during the technological process and by irradiation of 11000 fuel rods fabricated between 1977 and 1986 for the reactor BOR 60. (orig.).

171
Decontamination of liquid-metal fast breeder reactor components for reuse: The French experience. Michaille, P.; Moroni, J.C.; Lambert, I. *Nuclear Technology (United States);* 93: No. 2, 147-157 (Feb 1991).

Decontamination of stainless steel liquid-metal fast breeder reactor components for reuse in France began with the decontamination of Rapsodie components. At that time, dilute phosphoric acid was used. To cope with additional irradiated components after Phenix came into operation, an extensive study was performed, which led to the selection of a procedure involving two baths. The first bath, alkaline permanganate (AP), is applied for 3 h; the second bath, sulfo-phosphoric acid (SP), is applied for 6 h, both at 60°C. Up to three cycles are repeated until the residual dose rate is sufficiently low. Eight intermediate heat exchangers (IHXs) and two primary pumps from Phenix were decontaminated using this method. This paper reports that because SP can pickle only a limited depth ($\sim 3 \mu m$), due to the passivation effect of phosphoric acid, and because of the waste treatment problems associated with phosphates, new solutions were explored. One possibility involves improvement of the AP-SP procedure: In the SPm procedure, the AP bath is omitted and the phosphoric concentration is reduced by a factor of 4. A second approach is the use of a new formula, called SECA, a mixture of maleic and citric acid used in reducing conditions (imposed by hydrazine). Since the Phenix and Superphenix waste treatment facilities are not designed to reprocess maleic-citric acid, only the SPm procedure has been used on reactor components. A low-contaminated IHX from Rapsodie served as a test benchmark, not only for the decontamination procedure, but also for the requalification criteria, before the SPm procedure was applied to a highly contaminated IHX from Phenix. Recent results are presented.

172
Preparation, characterisation and out-of-pile property evaluation of (U,Pu)N fuel pellets. Ganguly, C.; Hegde, P.V.; Sengupta, A.K. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 234-241 (Feb 1991). (CONF-900625--).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

$(U_{0.45}Pu_{0.55})N$ and $(U_{0.8}Pu_{0.2})N$ are being considered in India as advanced alternative fuels for the operating fast breeder test reactor (FBTR) and the forthcoming prototype fast breeder reactor (PFBR). Mixed nitride fuel pellets containing <0.1 wt% each of oxygen and carbon impurities were fabricated by the conventional 'powder-pellet' (POP) and the advanced 'sol-gel microsphere pelletisation' (SGMP) processes, involving two major steps. First, carbothermic reduction of an oxide-graphite powder mixture (in the form of tablets) or gel-microspheres at 1773 - 1823 K in N_2 followed by N_2+H_2 and $Ar+H_2$ atmospheres. The nitride microspheres could be directly pelletised and sintered to pellets of relatively low density ($\leq$ 85% TD) with an 'open' pore structure desirable for LMFBR application. Thermal conductivity and hot hardness of nitride pellets were evaluated up to 1800 and 1500 respectively. The out-of-pile chemical compatibility experiments of mixed nitride fuel pellets for FBTR with SS 316 cladding at 973 K for 1000 h did not reveal any significant fuel-cladding chemical interaction. (orig.).

173
Post-buckling behavior during earthquakes and seismic margin of FBR main vessels. Hagiwara, Y.; Sawada, Y.; Akiyama, H.; Kokubo, K. *International Journal of Pressure Vessels and Piping (United Kingdom);* 45: No. 3, 259-271 (1991).

Shaking-table tests of cylindrical shells were performed in order to examine the buckling and post-buckling characteristics of Fast Breeder Reactor (FBR) main vessels under seismic shear loads. Static buckling tests were also performed, and it is confirmed that there is no significant difference between the static and dynamic load-displacement relations. Based on the test results, hysteresis rules of restoring force were formulated for both elastic and plastic shear-buckling. Non-linear dynamic-response analyses of the single-degree-of-freedom (SDOF) system were then carried out by using the hysteresis rules. The analyses were able to simulate the dynamic test results, especially energy-absorption capacity due to hysteresis behavior. Finally, the non-linear SDOF analysis was applied to the FBR main-vessel cylinder. It is pointed out that the hysteresis behavior could absorb a considerable amount of energy input from seismic motion, which would contribute to the seismic margin of FBR main vessels. (author).

174

Material redistribution in falling bundle structures under simulated severe FBR accident conditions. Peppler, W.; Will, H. *Nuclear Engineering and Design (Netherlands);* 126: No. 3, 403-412 (May 1991).

The course of severe FBR accidents, such as a single subassembly meltdown or disruptive whole-core accidents, is predominantly determined by the material motions and redistributions. In experiments using thermite to simulate the fuel and the chemical energy to simulate nuclear heat production the material redistribution was investigated in bundles undergoing meltdown. The use of X-ray cinematography allowed the observation of transient material movements. The fine dispersion of the material in the initiating phase, when driven by a steep pressure gradient, led to downstream material expulsion, which has been roughly quantified in the tests. Subsequent material flow resulted in blockage build-up in the simulated breeding zones. Three modes of hexcan failure were identified. The results of the SIMBATH test series (SIMBATH, simulation experiments in fuel element mock-ups with thermite) will be compared with those of in-pile experiments. (orig.).

Auxiliary, Mobile, Package, and Transportable

175

(N–91-27212)

Conference on Advanced Space Exploration Initiative Technologies. English, R.E. (National Aeronautics and Space Administration, Cleveland, OH (United States). Lewis Research Center). 1991. 14p. (NASA-TM–104527; E–6401; NAS–1.15:104527; AIAA–91-3562; CONF-9109226–). NTIS HC/MF A03; INIS.

From AIAA/NASA/OAI conference on advanced space exploration initiative (SEI) technologies; Cleveland, OH (United States) (3-4 Sep 1991).

Cosponsored by AIAA and OAI.

In striving to reduce exploration cost and exploration risks, a crucial aspect of the plans is program continuity, i.e., the continuing application of a given technology over a long period so that experience will accumulate from extended testing here on Earth and from a diversity of applications in space. An integrated view needs to be formed of the missions SEI will carry out, near term as well as far, and of the ways in which these missions can mutually support one another. Near term programs should be so constituted as to provide for the long term missions both the enabling technologies and the accumulation of experience they need. In achieving this, missions in Earth orbit should both evolve and show the technologies crucial to long term missions on the lunar surface, and the program for the lunar labs should evolve and show the enabling technologies for exploration of the surface of Mars and for flights of human beings to Mars and return. In the near term, the program for the Space Station should be directed and funded to develop and demonstrate the solar Brayton power plant that will be most useful as the power generator for the SP-100 nuclear reactor.

176

(SAND–91-1580C)

A charged particle transport analysis of the dose to a silicon-germanium thermoelectric element due to a solar flare event. Dandini, V.J. (Sandia National Labs., Albuquerque, NM (United States)). [1991]. Contract AC04-76DP00789. 4p. (CONF-910104–8). OSTI; NTIS; GPO Dep. Order Number DE91016650.

From 14. energy-sources technology conference and exhibition; Houston, TX (United States) (20-23 Jan 1991).

A version of the BRYNTRN baryon transport code written at the NASA Langley Research Center has been used to analyze the dose to a typical space reactor thermoelectric (TE) element due to a solar flare event. The code has been used in the past to calculate the dose/dose equivalent distributions to astronauts due to solar flares. It has been modified to accommodate multiple layers of spacecraft and component material. Differential and integrated doses to the TE element are presented and discussed. 5 refs.

177

The compatibility of urania and thoria with uranium tetrafluoride in the liquid and gas phase. Hanrahan, R.J. Jr.; Anghaie, S. pp. 29, Paper NUCL 98 of American Chemical Society, Division of Nuclear Chemistry and Technology. Washington, DC (US); American Chemical Society (1991). 34p. (CONF-910402–).

From 201. American Chemical Society (ACS) national meeting; Atlanta, GA (United States) (14-19 Apr 1991).

Work of the Innovative Nuclear Space Power and Propulsion Institute (INSPI) at the University of Florida, is directed at finding suitable materials for use in contact with uranium tetrafluoride from approximately 1,200 to 3,000C. Experiments were conducted on thorium dioxide (ThO_2) and uranium dioxide. Samples were exposed to liquid UF_4 at 1,100 C and to UF_4 vaporized at above 1,450C. At the conclusion of each exposure samples of residual gases diluted with nitrogen were run through a gas chromatograph (GC) to determine which gases were released as corrosion products. Powder samples of the surface scales and the bulk samples were then prepared for X-ray diffraction analysis (XRD) to determine composition. Surface analysis of the samples was conducted using Scanning Electron Microscopy (SEM) and Energy Dispersive X-ray Spectroscopy (EDS). Experiments with uranium dioxide showed that although UO_2 does not react significantly with UF_4, it does dissolve in liquid UF_4 and apparently suffers from ablation when exposed to UF_4 vapor. Thoria did react with UF_4 in both the liquid and gas phase exposures, forming a mixture of uranium dioxide and uranium-thorium oxyfluorides.

178

Nuclear power in space. Aftergood, S.; Hafemeister, D.W.; Prilutsky, O.F.; Rodionov, S.N.; Primack, J.R. *Scientific American (United States);* 264: No. 6, 42-47 (Jun 1991).

Nuclear reactors have provided energy for satellites-with nearly disastrous results. Now the US government is proposing to build nuclear-powered boosters to launch Star Wars defenses. These authors represent scientific groups that are opposed to the use of nuclear power in near space. The authors feel that the best course for space-borne reactors is to ban them from Earth orbit and use them in deep space.

179

A fission trip to Mars. Womack, S. *Engineer (London) (United Kingdom);* 272: No. 7050, 28-29 (9 May 1991).

Nuclear-powered rockets could be the driving force behind a new era of space exploration. The technical problems are briefly examined and found to be considerable. It is unrealistic to expect any fast breakthroughs in the technology. (author).

180
Thermal stress analyses of the multilayered fuel particles on a particle-bed reactor. Dobranich, D.; El-Genk, M.S. *Nuclear Technology (United States);* 94: No. 3, 372-382 (Jun 1991).

In this paper particle-bed reactors have been proposed to provide high-temperature, low-mass power sources for space-based operation. A computer program was prepared to simulate the thermal and mechanical response of a multilayered fuel particle operating in such a reactor. Issues of concern include temperature gradient and interference thermal stresses, along with the plastic and creep deformations associated with the high temperature of operation. The results of the computer simulations indicate that the interference thermal stress is much larger than the temperature gradient stress and the external pressure stress, and that permanent strain formation cannot be avoided for particles operating at temperatures greater than ~2300 K. The results also reveal some interesting aspects unique to multilayered fuel particle performance. Two such aspects include the interaction between interference thermal stress and high-temperature creep and the effect of power ramp time on the formation of time-dependent plastic strains.

RESEARCH, TEST AND EXPERIMENTAL REACTORS

181
Homogenization technique for strongly heterogeneous zones in research reactors. Lee, J.T.; Lee, B.H.; Cho, N.Z.; Oh, S.K. *Nuclear Technology (United States);* 94: No. 3, 286-296 (Jun 1991).

This paper reports on an iterative homogenization method using transport theory in a one-dimensional cylindrical cell model developed to improve the homogenized cross sections fro strongly heterogeneous zones in research reactors. The flux-weighting homogenized cross sections are modified by a correction factor, the cell flux ratio under an albedo boundary condition. The albedo at the cell boundary is iteratively determined to reflect the geometry effects of the material properties of the adjacent cells. This method has been tested with a simplified core model of the Korea Multipurpose Research Reactor. The results demonstrate that the reaction rates of an off-center control shroud cell, the multiplication factor, and the power distribution of the reactor core are close to those of the fine-mesh heterogeneous transport model.

182
Fission-product burnup chain model for research reactor application. Kim, Jung Do; Gil, Choong Sup; Lee, Jong Tai. *Journal of the Korean Nuclear Society (Wonjaryok Hakhoeji) (Korea, Republic of);* 22: No. 4, 351-358 (Dec 1990).

A new fission-product burnup chain model was developed for use in research reactor analysis capable of predicting the burnup-dependent reactivity with high precision over a wide range of burnup. The new model consists of 63 nuclides treated explicitly and one fissile-independent pseudo-element. The effective absorption cross sections for the preudo-element and the preudo-element yield of actinide nuclides were evaluated in the this report. The model is capable of predicting the high burnup behavior of low-enriched uranium-fueled research reactors.(Author).

PRODUCTION REACTORS

183
(ECN-I-91-026)
Safety aspects of HFR technical operation. Tas, A. (ed.). (Netherlands Energy Research Foundation, Petten (Netherlands)). Apr 1991. 24p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603394.

This annual report reports on the safety aspects of the technical operation of the High Flux Reactor, Petten during 1989. (H.W.). 4 figs.; 3 tabs.

184
Irradiation facilities development at CEA for solid and liquid breeder blankets qualification programs. Alberman, A.; Lefevre, F.; Thevenot, G.; Masson, M.; Savineau, M. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 179-181: No. pt.B, 1154-1157 (Mar-Apr 1991). (CONF-891204--).

From 4. international conference on fusion reactor materials; Kyoto (Japan) (4-8 Dec 1989).

Materials testing reactors of the Commissariat a l'energie atomique (CEA) have been involved in fusion technology materials qualification since 1983. At the earliest stage, tritium release as well as radiation damage studies on lithium ceramics were carried out in SILOE and OSIRIS reactors, relevant to NET breeder blanket design. In these programs, classical irradiation rigs were implemented, coping with a large number of small size specimens; they are still considered for ongoing programs. General layout was presented at the ICFRM-3 conference. Collaborative efforts of European experts groups to define NET breeder requirements have led CEA Prototype and Research Reactors Division to propose suitable irradiation facilities derived from its experience in fission reactors technology. Examples of breeder modules foreseen programs are: Pressurized water loops tests for Pb-17Li liquid module qualification for NET featuring on-line tritium monitoring in OSIRIS. Helium loop tests, derived from earlier HTGR fuel studies for solid blanket (ceramic-beryllium) qualification in SILOE for NET. DEMO relevant end-of-life tests on solid blanket in the PHENIX fast breeder. (orig.).

PROPULSION REACTORS

185
(ANL/CP-74388)
Development of nuclear fuels and materials for propulsion systems for SEI. Bhattacharyya, S.K.; Olsen, C.S.; Titran, R.H. (Argonne National Lab., IL (United States)). [1991]. Contract W-31109-ENG-38. 10p. (CONF-9109226--21). OSTI; NTIS (US Sales Only); GPO Dep. Order Number DE92000367.

From AIAA/NASA/OAI conference on advanced space exploration initiative (SEI) technologies; Cleveland, OH (United States) (3-4 Sep 1991).

Nuclear propulsion has been identified as an enabling technology in meeting the missions of the Space Exploration Initiative (SEI). Both Nuclear Thermal Propulsion (NTP) and Nuclear Electric Propulsion (NEP) have roles to play in the initiative. Of the numerous specific development items that need to be undertaken for these technologies, nuclear fuels and materials are considered by experts as the most challenging and therefore requiring early attention. One of the six panels organized by the NASA/DOE/DoD Steering Committee to help plan the nuclear propulsion development tasks was dedicated to nuclear fuels, materials and related technologies. The Panel was composed of volunteer representatives from most of the DOE laboratories and from NASA and DoD laboratories. In addition, industry "observers" provided significant input to the Panel deliberations. The Panel used the concepts presented at the 1990 Workshop on NTP and NEP as the bases for its fuels and materials development tasks. Since the goal is to achieve Technology Readiness Level (TRL) 6 by 2006, only the solid core concepts were considered in detail. It was concluded that there were classes of fuels and materials that had the potential to meet the demanding temperature and lifetime requirements for SEI. Plans for the development of fuels and materials were prepared. A full development plan is going to involve the construction of several major new facilities and the modification of many existing ones. For the fuels, areas of commonality have been explored to allow for effective early activity while the downselection process to focus the development takes place. For materials, the lack of definition of candidate selection by the concept proposers makes this somewhat more difficult. Careful coordination of the work will be essential to keep the development and characterization of fuels and materials on schedule and costs within bounds. 2 refs., 3 figs., 8 tabs.

186
(BNL-46294)
Assessment of the use of H_2, CH_4, NH_3 and CO_2 as NTR propellants. Selcow, E.C.; Davis, R.; Perkins, K.; Ludewig, H.; Cerbone, R.J. (Brookhaven National Lab., Upton, NY (United States)). [1991]. Contract AC02-76CH00016. 8p. (CONF-920104-7). OSTI; NTIS (US Sales Only); GPO Dep. Order Number DE91014794.

From 9. symposium on space nuclear power systems; Albuquerque, NM (United States) (13-16 Jan 1992).

In this paper the effect of changing from the traditional Nuclear Thermal Rocket (NTP) coolant, hydrogen, to several alternative coolants were studied. Hydrogen is generally chosen as an NTP coolant, since its use maximizes the specific impulse for a given operating temperature. However, there are situations in which it may not be available or optimal. The alternative coolants which were considered are ammonia, methane, and carbon dioxide. A particle bed reactor (PBR) generating 200 MW and cooled by hydrogen was used as the baseline against which all the comparisons were made. Both 19 and 37 elements cores were considered. The larger number of elements was found to be necessary in the case of carbon dioxide. The coolant reactivity worth was found to be directly proportional to the hydrogen coolant content. It was found that due to difference in the thermophysical proportions of the coolant that it would not be possible to use one reactor for all the coolants. The reactor would have to be constructed specifically for a coolant type. 1 ref., 1 fig., 4 tabs.

187
(LA-UR-91-3201)
Nuclear thermal rocket clustering: II, Monte Carlo analyses of neutronic, thermal, and shielding effects. Houts, M.G.; Buksa, J.J. (Los Alamos National Lab., NM (United States)). [1991]. Contract W-7405-ENG-36. 7p. (CONF-920104-12). OSTI; NTIS (US Sales Only); GPO Dep. Order Number DE92002366.

From 9. symposium on space nuclear power systems; Albuquerque, NM (United States) (13-16 Jan 1992).

Monte Carlo analyses of a cluster of nuclear thermal rockets (NTRs) have been performed using the code MCNP (Briesmeister 1986). Three effects of clustering have been analyzed: neutronic coupling, nuclear heating in a shutdown engine, and radiation scattering. Preliminary results indicate that while clustering effects should be taken into account, they will not have a significant effect on engine design or operation. More detailed analyses should be performed as engine and vehicle design progresses, although the basic conclusions of this study are not expected to change. 3 refs.

188
(LA-UR-91-3295)
Nuclear thermal rocket clustering: 1, A summary of previous work and relevant issues. Buksa, J.J.; Houts, M.G. (Los Alamos National Lab., NM (United States)). 14 Jul 1991. Contract W-7405-ENG-36. 10p. (CONF-920104-13). OSTI; NTIS; INIS; GPO Dep. Order Number DE92002374.

From 9. symposium on space nuclear power systems; Albuquerque, NM (United States) (13-16 Jan 1992).

A general review of the technical merits of nuclear thermal rocket clustering is presented. A summary of previous analyses performed during the Rover program is presented and used to assess clustering in the context of projected Space Exploration Initiative missions. A number of technical issues are discussed including cluster reliability, engine-out operation, neutronic coupling, shutdown core power generation, shutdown reactivity requirements, reactor kinetics, and radiation shielding. 7 refs., 3 figs., 2 tabs.

189
(N-91-27216)
An historical perspective of the NERVA nuclear rocket engine technology program. Final Report. Robbins, W.H.; Finger, H.B. (Analytical Engineering Corp., North Olmsted, OH (United States)). Jul 1991. 14p. (NASA-CR-187154; NAS-1.26:187154; AIAA-91-3451; CONF-9109226-). NTIS HC/MF A03; INIS.

From AIAA/NASA/OAI conference on advanced space exploration initiative (SEI) technologies; Cleveland, OH (United States) (3-4 Sep 1991).

Sponsored in part by NASA, AIAA, and OAI.

Nuclear rocket research and development was initiated in the United States in 1955 and is still being pursued to a limited extent. The major technology emphasis occurred in the decade of the 1960s and was primarily associated with the Rover/NERVA Program where the technology for a nuclear rocket engine system for space application was developed and demonstrated. The NERVA (Nuclear Engine for Rocket Vehicle Application) technology developed twenty years ago provides a comprehensive and viable propulsion technology base that can be applied and will prove to be valuable for application to the NASA Space Exploration Initiative (SEI). This paper, which is historical in scope, provides an overview of the conduct of the NERVA Engine Program, its organization and management, development philosophy, the engine configuration, and significant accomplishments.

190
(N-91-28193, pp. 431-449)
Nuclear thermal propulsion. Bennett, G.L. May 1991. vp. NTIS HC/MF A99; INIS. (NASA-CP-3112-VOL-2; NAS-1.55:3112-VOL-2; CONF-9006384-).

From Space transportation propulsion technology symposium; State College, PA (United States) (25-29 Jun 1990).

In Space Transportation Propulsion Technology Symposium. Volume 2: Symposium proceedings.

This document is presented in viewgraph form, and the topics covered include the following: (1) the direct fission-thermal propulsion process; (2) mission applications of direct fission-thermal propulsion; (3) nuclear engines for rocket vehicles; (4) manned mars landers; and (5) particle bed reactor design.

191
(N-91-28235)
Space Transportation Propulsion Technology Symposium. Volume 3: Panel Session Summaries and Presentations. (National Aeronautics and Space Administration, Washington, DC (United States)). May 1991. vp. (NASA-CP-3112-VOL-3; NAS-1.55:3112-VOL-3; CONF-9006384-). NTIS HC/MF A99; INIS.

From Space transportation propulsion technology symposium; State College, PA (United States) (25-29 Jun 1990).

The Space Transportation Propulsion Technology Symposium was held at the Pennsylvania State University on June 25 to 29, 1990. Emphasis was placed on propulsion requirements and initiatives to support current, next generation, and future space transportation systems, with the primary objectives of discerning whether proposed designs truly meet future transportation needs and identifying possible technology gaps, overlaps and other programmatic deficiencies. Key space transportation propulsion issues are addressed through four panels with government, industry, and academia membership. The panel focused on systems engineering and integration; development, manufacturing, and certification; operational efficiency; program development; and cultural issues.

192
(PNL-SA-29887)
PEGASUS: An integrated power and propulsion system for the space exploration initiative. Coomes, E.P.; Dagle, J.E. (Pacific Northwest Lab., Richland, WA (United States)). Sep 1991. Contract AC06-76RL01830. 14p. (CONF-9109226-19). OSTI; NTIS (US Sales Only); GPO Dep. Order Number DE92000145.

From AIAA/NASA/OAI conference on advanced space exploration initiative (SEI) technologies; Cleveland, OH (United States) (3-4 Sep 1991).

The advantages of using electric propulsion are well known in the aerospace community. The high specific impulse and, therefore, lower propellant requirements, make it a very attractive option for the Space Exploration Initiative (SEI). Recent studies have shown that nuclear electric propulsion (NEP) is not only attractive for the transport of cargo, but that fast-piloted missions to Mars are possible as well, with alphas on the order of 7.5 kg/Kw. An advanced NEP system with a specific power (alpha) of 2.5 kg/kW or less would significantly enhance the manned mission option of NEP by reducing the trip time even further. This paper describes an advanced system that combines the PEGASUS Drive with systems of the Rotating Multimegawatt Boiling Liquid Metal (RMBLR) power system that was developed as part of the US Department of Energy's (DOE) multimegawatt program, and was just recently declassified. In its original configuration, the PEGASUS Drive was a 10-MWe propulsion system. The RMBLR was a 20-MW electric power system. By combining the two, a second-generation PEGASUS Drive can be developed with an alpha of less than 2.5 kg/kW. This paper will address the technology advancements incorporated into the PEGASUS Drive, the analysis of a fast-piloted and unmanned cargo transport Mars mission, and the integration of laser power beaming to provide surface power. 15 refs., 3 figs., 5 tabs.

193
(SAND-91-1558C)
A fission fragment reactor concept for nuclear space propulsion. Suo-Anttila, A.J.; Parma, E.J.; Wright, S.A.; Vernon, M.E.; Pickard, P.S. (Sandia National Labs., Albuquerque, NM (United States)). [1991]. Contract AC04-76DP00789. 5p. (CONF-920104-9). OSTI; NTIS (US Sales Only); GPO Dep. Order Number DE91018000.

From 9. symposium on space nuclear power systems; Albuquerque, NM (United States) (13-16 Jan 1992).

Sandia National Laboratory (SNL) has proposed a new nuclear thermal propulsion concept that uses fission fragments to directly heat the propellant up to 1000K or higher above the material temperatures. The concept offers significant advantages over traditional solid core nuclear rocket concepts because of higher propellant exit temperatures while at the same time providing for more reliable operation due to lower structure temperatures and lower power densities. The concept can be operated in either steady state or pulsed modes. The engine consists of tubular modules, each with its own pressure boundary and rocket nozzle. The steady state mode requires a large engine with a reflector for criticality, provides high thrust and high ISP. The pulse mode utilizes a driver reactor for criticality, can be considerably smaller with lower but scaleable thrust. The pulse mode does require an external heat radiator for reactor cooling, which limits its duty cycle.

THEORY AND CALCULATION

194
(BARC-1543)
Monali-Rev.1: a Monte Carlo code for analysing fuel assemblies of nuclear reactors. Gupta, H.C. (Bhabha Atomic Research Centre, Bombay (India)). 1991. 56p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603256.

MONALI-Rev.1 is a multigroup Monte Carlo program developed on ND computers for analysing fuel assemblies of nuclear reactors. This version of the code is flexibly dimensioned so that the allowed size of a problem is limited only by the total data storage required. The code can read multigroup data for various nuclides directly from WIMS multigroup (69/27) cross section sets. Most of the input data, with the exception of cross sections, if needed, are read in free format. The treatment of anisotropy (up to P1 at present) may be in selective mixtures. The input to the geometry module has been simplified. The code has flexibility in the definition of regions. The results calculated by the code include K_{eff}, multigroup leakages and absorptions, group- and region-dependent fluxes. The multigroup leakages are calculated for each outer-most surface. Statistical confidence limits are also assigned to the results. In the end frequency distributions are found for the multiplication factor and optionally a normality test is also performed on the multiplication factors. (author). 8 refs., 4 figs., 2 tabs., 2 appendices.

195
(CONF-910603–22)
A technique for code validation for criticality safety calculations. Dyer, H.R.; Jordan, W.C.; Cain, V.R. (Oak Ridge National Lab., TN (United States)). [1991]. Contract AC05-84OR21400. 22p. OSTI; NTIS; GPO Dep. Order Number DE92002320.

From Annual meeting of the American Nuclear Society (ANS); Orlando, FL (United States) (2-6 Jun 1991).

There are probably as many techniques to validate computer codes for criticality safety purposes as there are computer codes and code validators. One method used at Martin Marietta Energy Systems, Inc., to validate the KENO code and associated cross sections consists of determining a single-sided, uniform-width, closed-interval, lower tolerance band for k_{eff} of critical systems. For application, this lower tolerance band becomes the upper safety limit (USL) acceptance criteria for subcriticality based upon the KENO calculations. A system is considered acceptably subcritical if a calculated k_{eff} plus 2 standard deviations lies below this upper safety limit (i.e., $k_{eff} + 2\sigma <$ USL). 5 refs., 1 fig.

196
(INDC(CCP)–336/L, pp. 119-138)
Proposal to represent neutron absorption by fission products by a single pseudo-fragment. Tsibulya, A.M.; Kochetkov, A.L.; Kravchenko, I.V.; Nikolaev, M.N. Aug 1991. 138p. Translated by A. Lorenz for the IAEA. Original report in Russian was distributed as INDC(CCP)-255/G. OSTI; NTIS (US Sales Only); INIS. Order Number DE92604085.

Translated by A. Lorenz for the IAEA. Original report in Russian was distributed as INDC(CCP)-255/G.

In Translation of selected papers published in Yadernye Konstanty (Nuclear Constants 4, 1985).

The concentration of fission products during reactor operation is analyzed. The dependence of a composite fission product capture cross-section as a function of time and on the nature of the A of the fissile nuclide are investigated, and the neutron radiative capture in fission products of a thermal reactor is evaluated. It is concluded that neutron absorption by fission products can be described by pseudo-fragments. (author). 18 refs, 2 figs, 3 tabs.

197
(ZJE–284)
Neutron transport planary slab-geometry shielding. Veverka, O. (Skoda, Plzen (Czechoslovakia). Zavod Vystavba Jadernych Elektraren). 1990. 115p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603257.

Discrete algebraic systems representing the Boltzmann kinetic equation for neutron transport within planar slab geometries were set up, including the boundary conditions and the continuity conditions on planes separating two different homogeneous materials. The kinetic equation was represented in the multigroup formulation in the PL, NPL, GLN, DSN and SN approximations, which are conventional in the treatment of the problem of multi-energy neutron fields and of the passage of neutrons through assemblies of materials. The formulation of the group constants is discussed in detail. A method for the calculation of the effective dose equivalent to man behind a biological shield of a nuclear reactor is also proposed. (author). 6 figs., 8 tabs., 12 refs.

198
Upper and lower bounds for disadvantage factors as a test of algorithm used in a synthesis method. Nanneh, M.M.; Ackroyd, R.T. *Aalam Al-Zarra (The World of Atom) (Syrian Arab Republic);* No. 14, 7-22 (Apr 1991). (In Arabic).

Translated from Annals of Nuclear Energy (1988) v. 15(5) p. 241-259.

A lower bound for the disadvantage factor of a lattice cell of arbitrary configuration is obtained using a finite element method which is based on a variational principle for the even-parity angular flux. An upper bound for the disadvantage factor is given by a finite element method using the complementary variational principle for the odd-parity angular flux. These theoretical results are illustrated by calculations for uranium/graphite and uranium/water lattices. As the approximations are refined the fluxes obtained by the first method tend towards the actual flux from below in the moderator, and from above in the fuel. These trends are reversed for the second method. This derivation of benchmarks for disadvantage factors has been undertaken primarily as a test of an important algorithm used by the authors in a method of synthesising transport solutions starting with a diffusion theory approximation. The algorithm is used to convert odd-parity approximations for the angular flux into even-parity approximations and vice versa. (author). 15 refs., 8 tabs., 9 figs.

199
A simple fuzzy simulation model for nuclear reactor system dynamics. Matsuoka, H. *Nuclear Technology (United States);* 94: No. 2, 228-241 (May 1991).

This paper presents a simple image model, the package flow model, for fuzzy simulation of nuclear reactor system dynamics. By using this model, fuzzy inference rules and their membership functions are easily obtained. The system dynamics can be approximately simulated by fuzzy inference. The method and some examples are described. The advantages of this model are intuitive understandability, flexible modification, and simplicity. Furthermore, high-speed calculation and high reliability can be realized by using fuzzy computing hardware in the near future.

200
A comparision of low-dimensional reactor kinetics analysis methods with modified Borresen's coarse-mesh method. Kim, Chang Hyo; Lee, Gyu Bok. *Journal of the Korean Nuclear Society (Wonjaryok Hakhoeji) (Korea, Republic of);* 22: No. 4, 359-370 (Dec 1990).

This study concerns with comparing low-dimensional reactor kinetics methods with a three-dimensional kinetics method to be used for safety analysis of light water reactors in order to suggest means of preparing input parameters required for low-dimensional methods. For this purpose a one-dimensional finite difference two-group diffusion theory code ODTRAN and a third-order Hermite polynomial-based point kinetics code POTRAN are developed and used to obtain low-dimensional solutions to the LRA-BWR kinetics benchmark problem. The results are compared with a three-dimensional modified Borresen's coarse-mesh solution of the kinetics problem by CMSNACK code. Through this comparision some simple but practical means of preparing input paramenters of low-dimensionsal kinetics analysis methods are suggested. (Author).

201
Analog and digital dynamic compensation techniques for delayed

self-powered neutron detectors. Yusuf, S.O.; Wehe, D.K. *Nuclear Science and Engineering (United States);* 106: No. 4, 399-408 (Dec 1990).

This paper reports on analog and digital methods developed to compensate for the time delay associated with rhodium self-powered neutron detector signals. This delay is caused by the decay of the neutron-activated rhodium and results in a current signal with unfavorable time response characteristics. The compensating analog method is based on the use of lead-lag networks to eliminate undesirable poles and zeros. The digital method takes digitized signals and numerically solves the inverse kinetics equation that relates reactor flux to the detector current at all earlier times. These methods were tested in a realistic reactor environment, and the results illustrate the accuracy achieved using each method.

202

Development of statistical core thermal design methodology using a modified latin hypercube method. Lee, S.H,; Kim, H.K.; Park, S.R.; Chang, S.H. *Nuclear Technology (United States);* 94: No. 3, 407-415 (Jun 1991).

This paper presents a statistical core thermal design methodology for generating the limit departure from nucleate boiling ratio (DNBR) and is used in assessing the best-estimate thermal margin in a reactor core. This new methodology adopts a modified Latin hypercube sampling method. In this method, the independencies of the input variables are verified through a correlation coefficient test for statistical treatment of their uncertainties. Next the DNBR response distribution is determined through a goodness-of-fit test. Finally, a limit DNBR with a one-sided 95% probability and a confidence level of 0.95 is estimated. This methodology is simpler than the conventional statistical method using the response surface and Monte Carlo simulation technique, but it maintains the same level of confidence in the limit DNBR result. From this study, it is deduced that the proposed methodology is useful for design application.

203

Thermal power prediction of nuclear power plant using neural network and parity space model. Roh, M.S.; Cheon, S.W.; Chang, S.H. *IEEE Transactions on Nuclear Science (Institute of Electrical and Electronics Engineers) (United States);* 38: No. 2, 866-872 (Apr 1991).

This paper reports on a power prediction system developed using an artificial neural network paradigm that was combined with a parity space signal validation technique. The parity space signal validation algorithm for the input preprocessing and the backpropagation network algorithm for the network learning are used for the power prediction system. A number of case studies were performed with emphasis on the applicability of the network in a steady-state high power level. The studies reveal that these algorithms can precisely predict the thermal power in a nuclear power plant. It also shows that the error signals resulting from instrumentation problems, even when the signals comprising various patterns are noisy or incomplete, can be properly treated.

COMPONENTS AND ACCESSORIES

204

(ECN-RX--91-015)

Fracture analysis of a pressure vessel rejected after inservice inspection. Braam, H.; Kan, L.J. van; Rongen, H.J.M. van. (Netherlands Energy Research Foundation, Petten (Netherlands)). Feb 1991. 8p. (CONF-910817--). OSTI; NTIS (US Sales Only); INIS. Order Number DE92603259.

From 11. international conference on structural mechanics in reactor technology; Tokyo (Japan) (18-23 Aug 1991).

As part of a research project aimed to validate fracture analysis methods as used for structures containing defects, a pressure vessel has been tested by pressurizing it till failure. The vessel (heat exchanger) was made available by DSM and was taken out of operation after defects had been found in the transition between the cylinder wall and the tube sheet of the vessel. Three different fracture analysis methods have been applied to predict the failure pressure of this vessel, viz. section XI of the ASME code, British Standards PD-6493 (1980) and CEGB R6-revision 3 (1986). To show the influence of the available information on the predicted critical load, 3 information levels have been defined varying from minimal to maximal information. The main difference between these levels involves the extent and the accuracy of the information necessary for the failure analysis. (author).3 refs.; 6 figs.

205

(EPRI-NP--7197)

Computer-assisted drawing information capture. Bhaskaran, P.; Filipski, A.; Flandrena, R.; Janney, M.; Ruggiero, S. (Electric Power Research Inst., Palo Alto, CA (United States); GTX Corp., Phoenix, AZ (United States)). Oct 1991. 40p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

This report documents the feasibility of automating most of the process for converting paper-based drawings into intelligent electronic form. Use of computerized automation tools could potentially save millions of dollars over existing manual drawing conversion methods. The project team visited five utility sites to investigate current design and drafting practices and costs. Next, they developed a produce description, system requirements, and a cost-benefit analysis. Prototypes demonstrated the feasibility of rule-based symbol recognition and use of neural networks to achieve high-recognition rates for hand-lettered text. Utility reviews provided feedback on the requirements, cost-benefit analysis, product description, and design specification. This report documents the feasibility of developing a computerized automation tool that performs symbol and character recognition using contextual information to improve recognition accuracy, and then it associates symbols and text. This report contains the product description, cost-benefit analysis, and feasibility prototype results. The cost-benefit analysis demonstrates that such a tool would provide significant savings and offer considerable advantages to utilities. Furthermore, feasibility prototypes demonstrate that even without the use of context information, rule-based recognition software could recognize 70% of the symbols, and neural network character recognition software couple recognize 90--100% of hand-lettered characters. 9 figs., 14 tabs.

206

(EPRI-NP--7410-Vol.3)

Breaker Maintenance: Volume 3, Molded-Case circuit breakers. Davis, E.L.; Funk, D.L. (Electric Power Research Inst., Palo Alto, CA (United States); Edan Engineering Corp., Portland, OR (United States)). Sep 1991.

95p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Modeled-case circuit breakers (MCCBs) provide power and circuit protection in nuclear plant electrical distribution systems. Thus, their proper operation is essential to the safe and reliable operation of plant electrical distribution systems. This guide for both nuclear and non-nuclear power generating facilities will help improve the maintenance and reliability of MCCBs. The authors developed this guide to establish a working-level understanding of hardware performance trends, reliability, and failure modes from which maintenance practices could be specified. The first step in preparing this guide was an in-depth review of available operating experience and failure data, which was obtained from nuclear information sources such as EPRI's Nuclear Maintenance Applications Center. In addition, they evaluated some nonnuclear reliability data. Next, they investigated current industry practices, including a review of manufacturer's recommendations and numerous industry standards. Finally, they used the collective information to develop programmatic recommendations and, where appropriate, detailed inspection and test guidance. This guide offers many recommendations applicable to MCCB maintenance, such as an engineering description of MCCBs and their operation; an overview of reliability and failure data; programmatic recommendations, including inspection and test periodicity; detailed inspection and test guidance; and corrective maintenance recommendations. Supplementary information includes an overview of industry standards, a discussion of regulatory issues, and sample data sheets. An evaluation of operating experience indicates that a maintenance program need not be overly complicated to detect the predominant failure modes associated with MCCBs. Those implementating an MCCB maintenance and testing program, however, must recognize the inherent limitations associated with field-testing of these devices. 42 refs., 17 figs., 7 tabs.

207
(EPRI-NP–7466-M)
Nondestructive evaluation sourcebook. Ammirato, F.V.; Walker, S.M.; Nottingham, L.D.; Stephens, H.; Shankar, R.; Krzywosz, K.; Gothard, M. (Electric Power Research Inst., Palo Alto, CA (United States); Jones (J.A.) Applied Research Co., Charlotte, NC (United States)). Sep 1991. 11p. Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

Utility executives and upper level managers often make decisions based on inspection data and opinions of inspection personnel regarding inservice inspections of critical components such as pressure vessels, piping, steam generators, and turbine-generator rotors. Few utility executives and upper level managers, however, are well versed in the non-destructive evaluation (NDE) technology that is applied in their nuclear plants. The capabilities and limitations of NDE technology, even though well established and documented for many applications, are not well known at the upper management level. The purpose of this sourcebook is to provide utility upper management and executives with information that explains how NDE is performed in their plants, how the NDE data is used, what training and qualifications are required for NDE personnel, and where and how to get more information. The sourcebook is not intended as an NDE textbook or training manual; its main objective, rather, is to provide an overview of NDE and to give the reader access to the wide selection of available, detailed information on NDE and its application in nuclear plants. Although the sourcebook addresses mainly nuclear plant NDE, much of the information is applicable to fossil plants. 6 refs.

208
(EPRI-NP–7523)
Proceedings: Instrumentation and control test reduction workshop. (Electric Power Research Inst., Palo Alto, CA (United States); Ordis, Inc., San Jose, CA (United States)). Sep 1991. 256p. (CONF-9101115–). Research Reports Center, PO Box 50490, Palo Alto, CA 94303.

From Instrumentation and control test reduction workshop; Monterey, CA (United States) (24-25 Jan 1991).

Instrumentation and control (I&C) surveillance and testing is a significant contributor to operations and management costs. Several techniques to eliminate or reduce manual testing requirements could reduce costs while improving plant safety and performance. I&C test reduction was the subject of this 1991 workshop. The workshop covered test elimination, test reduction, test automation, and relevant standards and benefits. The conclusions of the workshop were the following: More utility information sharing is important. There is a significant amount of information available throughout the industry, but it is not available in a concise, useable form. An I&C utility users group is needed to address items such as instrument calibration reduction, set-point methodologies, and other current I&C issues. The workshop was well received. The timing is right to initiate actions to reduce testing.

209
(KAPL–4725)
Stress intensity factors for an underclad nozzle corner crack subjected to pressure and thermal loading. Wilkening, W.W. (Knolls Atomic Power Lab., Schenectady, NY (United States)). Jun 1991. 27p. (CONF-910602–51). Distribution: UC-504. OSTI; NTIS; GPO Dep. Order Number DE92000492.

From American Society of Mechanical Engineers (ASME) pressure vessels and piping conference; San Diego, CA (United States) (23-27 Jun 1991).

The opening mode linear elastic stress intensity factor, K_I, was computed, via 3-D elastic finite element techniques, for an embedded elliptical crack located just beneath the cladding at the nozzle corner in a pressure vessel. Pressure loading and several thermal transient loading conditions were analyzed. The underclad crack was explicitly modeled and K_I was computed explicitly, from the energy release rate, J. The variation of the maximum principal stress along the minor axis of the elliptical crack was determined for a companion set of thermal/structural analyses that were performed in the absence of the crack. These stress distributions were linearized into equivalent membrane and bending stress components that were used to compute K_I from the Shah and Kobayashi solutions for near-surface embedded elliptical cracks. The explicitly computed K_I values were found to be in very good agreement with the K_I values computed from the "flat plate" solutions of Reference 1, for all the loading cases analyzed. An additional comparison was made between the energy release rate results and the results obtained by fitting the $1/\sqrt{r}$ stress singularity to the crack tip stress field at the Gaussian integration points nearest to the crack front. The observed excellent agreement between the two independent "explicit" computational methods served to verify each of the methods and also demonstrated the adequacy of the refinement of the finite element mesh. These observations support the use of the Shah and Kobayashi flat plate K_I solutions for analyzing underclad cracks at the nozzle corner. 7 refs., 11 figs.

210
(NUREG/CR-5651)
An investigation of crack-tip stress field criteria of predicting cleavage-crack initiation. Keeney-Walker, J.; Bass, B.R.; Landes, J.D. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering; Oak Ridge National Lab., TN (United States)). Sep 1991. Contract AC05-84OR21400. 34p. (ORNL/TM-11692). OSTI; NTIS; INIS; GPO.

Cleavage-crack initiation in large-scale wide-plate (WP) specimens could not be accurately predicted from small, compact (CT) specimens by using a linear-elastic fracture-mechanics, K_{Ic}, methodology. In the wide-plate tests conducted by the Heavy-Section Steel Technology Program at Oak Ridge National Laboratory, crack initiation has consistently occurred at stress-intensity (K_I) values ranging from two to four times those predicted by the CT specimens. Studies were initiated to develop crack-tip stress field criteria incorporating effects of geometry, size, and constraint that will lead to improved predictions of cleavage initiation in WP specimens from CT specimens. The work centers around nonlinear two-and three-dimensional finite-element analyses of the crack-tip stress fields in these geometries. Analyses were conducted on CT and WP specimens for which cleavage initiation fracture had been measured in laboratory tests. The local crack-tip field generated for these specimens were then used in the evaluation of fracture correlation parameters to augment the K_I parameter for predicting cleavage initiation. Parameters of hydrostatic constraint and of maximum principal stress, measured volumetrically, are included in these evaluations. The results suggest that the cleavage initiation process can be correlated with the local crack-tip fields via a maximum principal stress criterion based on achieving a critical area within a critical stress contour. This criterion has been successfully applied to correlate cleavage initiation in 2T-CT and WP specimen geometries. 23 refs., 16 figs., 5 tabs.

211
Nuclear safety - the EC perspective. Finzi, Sergio. *Physics World (United Kingdom);* 4: No. 8, 43-45 (Aug 1991).

A European cooperative approach to the safe use of nuclear power is the avowed strategy of the European Economic Community. A resolution passed in 1975 required the harmonization of safety standards for nuclear installations, systematic evaluation of those installations, continuous efforts to improve safety using international cooperation. This article examines the degree to which these criteria have been met. Accident simulations and risk assessment are both being used to meet the safety requirements. (UK).

212
Planning for protecting power plants must not be just an afterthought. Hogg, Bruce. *Fire (1908amp) (United Kingdom);* 84: No. 1033, 21-22 (Jul 1991).

In this paper the Principal Fire and Major Incident Engineer at Nuclear Electric plc looks at fire safety in nuclear power plants, from the application of sophisticated 'state of the art' detection and extinguishment systems to the provision of well-trained firefighting teams. (author).

213
A different approach to quantifying fire risks when decision making. Hay, Adrian. *Fire (1908amp) (United Kingdom);* 84: No. 1033, 32, 34 (Jul 1991).

This article outlines an alternative approach to fire safety design which involves decision making on the basis of risk considerations. The methodology is being developed in conjunction with the nuclear industry but is considered equally applicable to other industrial facilities. (author).

214
Grooved tube plug rolls in. Krausser, P. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 28, 30 (Jul 1991).

The removable plugs used to date by the Power Generation Group (KWU) of Siemens to seal defective steam generator tubes have a good track record. Their sealing principle is based on the elastic tensioning of three seal disks against the inside wall of the tube. Now a further removable plug is available -a roll-in plug with a metal-coated surface. It is particularly suitable for use in the roller-expanded zone of the tubes at the tube sheet. The plugs can be used in both Siemens-KWU steam generators and in steam generators manufactured in compliance with the guidelines of the ASME Code. (author).

215
Is nuclear power over-engineered?. Hay, K. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 444, 37-39 (Jul 1991).

Recent developments in the understanding of many aspects of nuclear power have revealed that safety margins sensibly built into nuclear systems ten or twenty years ago may in fact be over-conservative to the point of actually hampering performance. There will continue to be areas where large margins are desirable, but there may be scope for some over-conservatisms to be removed so that each individual part can operate up to its sensible and safe limit. Determining realistically what these limits are is the important thing. Areas discussed here include criticality codes, thermal hydraulics, management of irradiated fuel, plant aging and structural integrity. (author).

216
The changing technology of air filtration. Weber, L.D. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 54-56 (Jun 1991).

Air cleaning in nuclear power plants is undergoing some of the most significant evolutionary changes since the late 1950s, when HEPA (High Efficiency Particulate Air) filter media were in transition from cellulosic to more heat and combustion-resistant glass fibre-based materials. HEPA filters are now widely specified for nuclear reactor, fuel and waste storage containment ventilators, radwaste and fuel processing off-gas treatment, and they are essential to operator and environmental safety in any process where air-borne contaminant particles may be generated. The limitations of these filters and problems encountered in their use are discussed. New regulations for filtration equipment are stimulating interest in new filter technology capable of meeting the more stringent standards required. Metal filters offer a promising alternative to glass fibre filters. (author).

217
Critical programme for nuclear safety. Wyman, V. *Engineer (London)*

(United Kingdom); 272: No. 7051, 36, 38 (16 May 1991).

This July should see the award of the first contracts under a Pound 25 million programme designed to lead to a better understanding of control systems for safety-critical uses. The research and development programme is aimed at safety-critical circuits in everything form electronically-controlled car suspensions to safety systems in nuclear power stations. This article looks specifically at the latter. (author).

218
Tearing stability analysis of an axial surface flaw in thick-walled pressure vessels. Zahoor, A.; Ghassemi, B.B. *Nuclear Engineering and Design (Netherlands);* 126: No. 1, 61-69 (Apr 1991).

This paper presents two fracture mechanics models for evaluation of an axial surface flaw in pressure vessels. The surface flaw is located on the outside surface of the vessel. The first model assumes yielding of the remaining ligament directly ahead of the flaw. The second model assumes contained yielding ahead of the flaw and uses a linear elastic fracture mechanics solution. The former model is suitable for cases where the combination of material toughness, flaw size, and load is such that initiation of flaw growth follows ligament yielding. The latter model is suitable for low-toughness materials where initiation of crack growth and potential tearing instability may occur prior to the yielding of the ligament. Both models are suitable for thick-walled vessels. The paper discusses the applicability regime for both models. The models are then applied to a test vessel and the predicted failure pressure is compared against the pressure attained in the test. Results show that both models can be applied successfully. In particular, the contained yielding model when used with the plane-stress assumption can give reasonable predictions even for cases that involve yielding of the ligament. (orig.).

219
The RETRAN-03 computer code. Paulsen, M.P.; McFadden, J.H.; Peterson, C.E.; McClure, J.A.; Gose, G.C.; Jensen, P.J. *Nuclear Technology (United States);* 93: No. 1, 105-115 (Jan 1991).

The RETRAN-03 code development effort is designed to overcome the major theoretical and practical limitations associated with the RETRAN-02 computer code. The major objectives of the development program are to extend the range of analyses that can be performed with RETRAN, to make the code more dependable and faster running, and to have a more transportable code. The first two objectives are accomplished by developing new models and adding other models to the RETRAN-02 base code. The major model additions for RETRAN-03 are as follows: implicit solution methods for the steady-state and transient forms of the field equations; additional options for the velocity difference equation; a new steady-state initialization option for computer low-power steam generator initial conditions; models for nonequilibrium thermodynamic conditions; and several special-purpose models. The source code and the environmental library for RETRAN-03 are written in standard FORTRAN 77, which allows the last objective to be fulfilled. Some models in RETRAN-02 have been deleted in RETRAN-03. In this paper the changes between RETRAN-02 and RETRAN-03 are reviewed.

220
Study on the automatization of the steam generator water level at low power. Lee, Yoon Hoon. *Cheju University Journal (Korea, Republic of);* 31: 143-151 (Dec 1990).

The water level control of the steam generator in a nuclear power plant is difficult to control at low power because of its reverse directional response to feedwater control. The main reason of this behavior is the intrinsic thermal-hydraulic properties of the steam generator which are well known as swell and shrink phenomena. These phenomena are common to all power plants and become more salient as plant power decreases. The failure of the water control causes reactor trips, most of which are spurious at low power. These spurious trips are not desirable with respect to both the plant availability as well as safety. A new control system is proposed in this paper for the automatic control of steam generator water level without interrupting stability and still with proper control characteristics. This control scheme, named off-line guider, is developed by the use of thermal-hydraulic characteristics of steam generator and responses to key parameters are expressed in the form of transfer functions. With these tranfer functions known and by selecting proper compositions and control constants, the control scheme for automatic control can be defined.(Author).

221
U-tube steam generator predictions: Bundle convective heat transfer correlations. Hassan, Y.A.; Kalyanasundaram, M. *Nuclear Technology (United States);* 94: No. 3, 394-406 (Jun 1991).

This paper reports on the development of a RELAP5/MOD2 computer code model for a Model Boiler-2 U-tube steam generator (UTSG) to predict the thermal-hydraulic response of a UTSG during steady-state operation and for a loss-of-feedwater (LOF) transient. Steady-state conditions calculated by RELAP5 are compared with the measured data. The calculated heat transfer from the primary to the secondary side of the steam generator is found to be underpredicted by 30%. The heat transfer correlations used in existing thermal-hydraulic codes are developed for flow inside individual tubes and not for flow around tube bundles. Consequently, the secondary convective heat transfer is not accurately predicted by the codes. A revised version of the RELAP5 code with modified heat transfer correlations reasonably predicts the primary to the secondary heat transfer in bundle environments. Improved heat fluxes and heat transfer coefficients are obtained during steady-state and LOF accident transients. Steady-state behavior of the Semiscale MOD-2C steam generator is also computed with both the original and the revised versions of the code. Good agreement is achieved between the predictions and the test data when the modified heat transfer correlations are utilized.

222
The influences of mesh subdivision on nonlinear fracture analysis for surface cracked structures. Shimakawa, T. *International Journal of Pressure Vessels and Piping (United Kingdom);* 45: No. 3, 327-349 (1991).

The leak-before-break (LBB) concept can be expected to be applied not only to safety assessment, but also to the rationalization of nuclear power plants. The development of a method to evaluate fracture characteristics is required to establish this concept. The finite element method (FEM) is one of the most useful tools for this evaluation. However, the influence of various factors on the solution is not well understood and the reliability has not been fully verified.

In this study, elastic-plastic 3D analyses are performed for two kinds of surface cracked structure, and the influence of mesh design is discussed. The first problem is surface crack growth in a carbon steel plate subjected to tension loading. A crack extension analysis is performed under a generation phase simulation using the crack release technique. Numerical instability of the J-integral solution is observed when the number of elements in the thickness direction of the ligament is reduced to three. The influence of mesh design in the ligament on the solution is discussed. The second problem is a circumferential part-through crack in a carbon steel pipe subjected to a bending moment. Two kinds of mesh design are employed, and a comparison between two sets of results shows that the number of elements on the crack surface also affects the solution as well as the number of elements in the ligament. (author).

223
Application of ultrasonic inspection data in strength calculations for nuclear power plant equipment. Ovchinnikov, A.V.; Rivkin, E.Yu.; Vasilchenko, G.S.; Zvezdin, Yu.I. *International Journal of Pressure Vessels and Piping (United Kingdom);* 45: No. 3, 357-365 (1991).

Several kinds of test specimens were produced with three types of defects of defined sizes and positions in the particular localities of weld joints. Such specimens have been used for defect parameter characterization by ultrasonic testing. The principles for schematization of such defects and the formulae for the stress intensity factor calculations for elliptical and semielliptical cracks have been worked out. Methods for defining the sizes of defect which are acceptable have been designed for use for use on operational nuclear power plant equipment and take account of the mutual effects of the force, thermal and residual stresses. The method can be used in the brittle, transitional and tough material state. (author).

224
Repair and completion of damaged cooling tower. Gould, P.L.; Guedelhoefer, O.C. *Journal of Structural Engineering (United States);* 115: No. 3, 576-593 (Mar 1989).

This paper reports on a large hyperbolic cooling tower, under construction and nearly completed, struck by a falling tower crane during a tornado. Damage occurred at the upper edge where a V-shaped notch was gouged. Also, considerable cracking beneath the notch was observed. The extent of the damage was documented by precision survey techniques and visual inspection. A comprehensive analytical study was performed to insure that the completed tower would meet the design criteria. The repair plan involved repairing the cracks, sawing back the notch in a step fashion, refurbishing the scaffolding, rebuilding the gouged region, and then carrying the construction to completion. Also, two circumferential stiffening rings were added to the shell.

225
On the steady-state performance of natural circulation loops. Vijayan, P.K.; Mehta, S.K.; Date, A.W. *International Journal of Heat and Mass Transfer (United Kingdom);* 34: No. 9, 2219-2230 (Sep 1991).

This paper reports a comparison between natural and forced circulation data in a figure-of-eight loop relevant to a pressure tube type heavy water reactor. It is shown that both friction and heat transfer are affected by the presence of buoyancy induced secondary flows under steady-state natural circulation conditions. Several past proposals for correlating the overall hydraulic loss coefficient are reviewed and a generalized correlation is proposed that is non-loop specific. The correlation successfully predicts experimental data from three different loops. The experimental data, however, are generated with seven different geometric configurations of these three loops. (author).

226
Apparatus for servicing a jet pump hold down beam in a nuclear reactor. Howell, D.A.; Hydeman, J.E.; Slater, J.L.; Bodnar, R.J.; Golick, L.R.; Sckera, R.S.; Roth, C.H. Jr. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). USA Patent 4,995,158/A/. 26 Feb 1991. Filed date 8 Feb 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes an apparatus for replacing the hold down beam of a fluid circulating jet pump mounted in a nuclear reactor, the hold down beam having a beam body, a pair of opposed beam tabs and a pair of opposed beam positioning trunnions extending outwardly from the beam body. It comprises a housing having a lower surface configured to be positionable over the body of the hold down beam; means coupled to the housing for engaging the beam trunnions and securing the beam body against the lower surface of the housing; means coupled to the housing for depressing the beam tabs while the beam body is secured against the lower surface of the housing; means coupled to the trunnion engaging means and the beam tab depressing means for selectively actuating the trunnion engaging means and the beam tab depressing means from a position remote from the nuclear reactor; and means connectable to the housing for selectively changing the directional orientation of the beam.

227
Core catchers for nuclear reactors. McIntyre, Micheal; Gardner, I.P. UK Patent 2236210/A/. 27 Mar 1991. 30p. Available from the Patent Office, Sales Branch, St. Mary Cray, Orpington, Kent BR5 3RD.

A core catcher for containing nuclear core debris in the event of a breach in the reactor pressure vessel caused by a core meltdown is described. It has a multilayer sandwich construction comprising a middle layer of interlocking tongue-and-groove jointed refractory (e.g. zirconia) tiles or bricks sandwiched between inner and outer steel plates in the form of domes. The refractory bricks are fixed against movement relative to each other and the inner and outer steel plates by means of refractory cement. The inner steel plate is sacrificial in the event that it comes into contact with molten nuclear material but gives the sandwich construction greater shock resistance during normal operational service. The outer steel plate provides the main structural support for the core catcher. (author).

228
Improvements in acoustic tomography. Olley, P.; MacLeod, I.D.; Beesley, M.J. UK Patent 2235294/A/. 27 Feb 1991. 6p. Available from The Patent Office, Sales Branch, St Mary Cray, Orpington, Kent BR5 3RD.

An array of acoustic transducers is arranged around an area of interest such as a water tank and each transducer in turn transmits pulses of sound signals which are received by each of the other transducers. The area is notionally sub-divided into a square grid cell being traversed by at least one sound path between transducers. The acoustic characteristics can be determined for each cell and used to measure temperature and/or to detect gas bubbles in a fluid from sound propagation velocity and attenuation. (author).

MATERIALS

229
(INIS-mf-13013)
Corrosion problems of power engineering. (Skoda, Plzen (Czechoslovakia). Zavodni Pobocka Ceske Vedeckotechnicke Spolecnosti; Vysoka Skola Strojni a Elektrotechnicka, Plzen (Czechoslovakia); Ceskoslovenska Akademie Ved, Plzen (Czechoslovakia). Ustav Teoreticke a Aplikovane Mechaniky). 1989. 151p. (In Czech, Slovak). (CONF-8912162–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92604650.

From 8. conference on corrosion problems in power engineering; Marianske Lazne (Czechoslovakia) (12-14 Dec 1989).

The proceedings contain 26 contributions, out of which 11 have been inputted in INIS. These are concerned with methods for the evaluation of corrosion resistance of materials for the nuclear industry, with examination of the corrosion behavior of composite overlays and of steels after the action of decontamination solutions, and with theoretical models of crack propagation. Corrosion problems of steam turbines, steam generator tubes and thermocouple bushings are discussed. (M.D.). 28 figs., 8 tabs., 63 refs.

230
Application of surface science to the study of the corrosion of PWR primary circuit materials. Harris, S.J. (Southampton Univ. (United Kingdom)). Apr 1989. 577p. Available from British Library Document Supply Centre, Boston Spa, Wetherby, West Yorks. LS23 7BQ. No. DX89585.

This thesis describes a study of the corrosion and oxidation of PWR primary circuit materials using surface sensitive spectroscopic techniques. An X-ray photoemission spectroscopy (XPS) study of a number of mixed oxides of known composition is described and the information obtained is related to XPS measurements made on the surface of iron and nickel based alloys oxidised under controlled conditions. A secondary ion mass spectroscopy (SIMA) study on these mixed transition metal oxides is also described. The gaseous oxidation of stainless steel 3041 and Inconel-690 is examined. Both alloys were oxidised at 600K in air with the composition of the oxide films formed studied by a range of surface spectroscopic methods. Further experimental work was performed on Inconel-690 to examine the effects of surface pretreatment and the effects of low oxygen partial pressures on the formation of oxide films at 600 K. The incorporation of the radionuclide, cobalt-60, into the oxide films formed on structural components of a PWR, result in the build up of radiation fields. A method of pretreating the surface of the alloy stainless steel 3041, in order to reduce the level of cobalt adsorbed into the oxide film formed under simulated primary coolant conditions is examined and contrasts with treatments which have been developed to release cobalt adsorbed in existing oxide layers under reactor conditions are discussed. (author).

231
An oxide-semiconductance model of nodular corrosion and its application to zirconium alloy development. Taylor, D.F. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 184: No. 1, 65-77 (Aug 1991).

The mechanisms of nodule initiation and growth on susceptible zirconium alloys both in- and ex-reactor remain unclear. Hypotheses citing locally inadequate concentrations of alloying elements generally can explain more experimental observations than, for example, those relying solely on electrochemical activity at second phase particles, or residual stress, or texture. Since the alloying additions that protect zirconium from rapid oxidation in high-temperature, high-pressure steam all have normal oxidation states other than IV, it seems likely that the incorporation of aliovalent oxides into the monoclinic zirconia lattice plays an important role in providing that protection. Corrosion experiments indicated that effective elements are those forming aliovalent cations that can substitute for Zr(IV) and alter the defect structure without disrupting the oxide matrix. Laboratory alloy optimization for a combination of immunity to nodular corrosion in steam and minimum uniform corrosion in water requires in-reactor confirmation. (orig.).

232
Influence of sodium on creep-rupture behaviour of type 304 stainless steel. Borgstedt, H.U.; Huthmann, H. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 183: No. 3, 127-136 (Aug 1991).

The influence of flowing liquid sodium at 550deg C on the creep-rupture behaviour of the structural material of the SNR 300 reactor, X6CrNi18 11 (DIN 1.4948, equivalent to Type 304 SS) was studied in two non-isothermal sodium loops. It was shown that the effects of sodium are dependent on the carbon activity of the sodium. Under normal (non-decarburizing) sodium conditions a limited reduction of times-to-rupture occurs. This reduction is due to a reduced ductility within the tertiary creep range. The minimum creep rate and the onset of tertiary creep are not influenced. Under decarburizing conditions, which are not expected to occur in the sodium of LMFBRS, an additional loss of creep strength was observed. The steel showed higher creep rates and an earlier onset of tertiary creep. This additional effect seems to be caused by sodium corrosion of surface-near layers reducing the unaffected cross section. It depends on the surface-to-volume ratio, and it was nearly suppressed, when thicker specimens (6 mm diameter) were used. (orig.).

233
A model for waterside oxidation of Zircaloy fuel cladding in pressurized water reactors. Almarshad, A.I.A.; Klein, A.C. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 183: No. 3, 186-194 (Aug 1991).

A model is developed to simulate the oxidation of Zircaloy fuel rod cladding exposed to pressurized water reactor operating conditions. The model is used to predict the oxidation rate for both ex- and in-reactor conditions in terms of the weight gain and oxide thickness. Comparisons of the model predictions with experimental data show very good agreement. (orig.).

234
Effect of sintering atmosphere on the densification of UO_2-Gd_2O_3 compacts. Yuda, Ryoichi; Une, Katsumi. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 195-203 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

Sintering kinetics of powder compacts of UO_2-(5,10 wt%) Gd_2O_3 and

UO_2 have been studied in controlled atmospheres of H_2O/H_2 and CO_2/CO mixed gases by using a dilatometer. The densification rates and microstructure of the sintered pellets are considerably influenced by both the sintering atmosphere and Gd_2O_3 content. After a heat treatment of 1650deg C for 2 h, the sintered densities for UO_2-Gd_2O_3 pellets begin to decrease above the threshold oxidizing atomospheres, while the density for the UO_2 pellet increases slightly with more oxidizing atmospheres. These behaviors result from the difference in development of pore structure during sintering: the pore structure of UO_2-Gd_2O_3 pellets varies from an open pore structure to a closed pore structure on changing the sintering atmosphere from reducing to oxidizing. On the other hand, the pore structure of the UO_2 pellet is hardly affected by the sintering atmosphere. The formation of $(U,Gd)O_2$ solid solutions and the grain growth are enhanced with more oxidizing atmospheres. (orig.).

235
Localized impedance-anodization measurements to characterize corrosion films on irradiated zirconium alloys. Ramasubramanian, N.; Ling, V.C. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 183: No. 3, 226-228 (Aug 1991).

Letter-to-the-editor.

Impedance and anodization measurements, using an aqueous electrolyte contact and localized to small areas, were used to grade the quality of the corrosion film on irradiated zirconium alloys. This method was developed for examining oxide films on the outside surface of pressure tubes removed from CANDU reactors. (orig.).

236
UO_2-Gd_2O_3 solid solution formation from wet and dry processes. Riella, H.G.; Durazzo, M.; Hirata, M.; Nogueira, R.A. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 204-211 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

Gadolinium oxide homogeneously mixed with uranium oxide nuclear fuel is used as burnable poison in modern LWR. Solid solutions of UO_2 with Gd_2O_3 additions of between 5 and 10 wt% have been prepared according to the following routes: (a) mechanical mixing of both ceramic powders; (b) AUC and ADU coprecipitation. Microstructural analyses have been carried out by optical ceramography and electron microprobe for observation of heterogeneities. Sintered densities as well as O/U ratios of the ceramic pellets have also been measured. (orig.).

237
Viscoelastic stress analysis of nonaxisymmetrically heated cylindrical tubes. Park, Jin Seok; Seo, Keum Seok; Na, Bok Gyun; Kim, Jong In. *Transactions of the Korean Society of Mechanical Engineers (Korea, Republic of);* 15: No. 2, 596-603 (Mar 1991). (In Korean).

A solution is presented for the computation of the elastic-creep stresses in a hollow cylinder subjected to nonaxisymmetric temperature distribution. The creep problem is treated by the Maxwell creep model. Laplace transformation is used for reformation of the governing equation of elastic problem and Hooke's law in a function of γ,θ,and creep constant. The governing equation is set up using the Airy stress function which leads to the biharmonic equation. The solution is obtained by using Fourier series method used to obtain the stress components which include the variation of time. This solution shows excellent agreement with Lamkin's and Boley and Weiner's solution. The viscoelastic stresses are also obtained for the fuel rod tube subjecting nonaxisymmetric thermal load. (Author).

238
Carbon monoxide-silicon carbide interaction in HTGR fuel particles. Minato, Kazuo; Ogawa, Toru; Kashimura, Satoru; Fukuda, Kousaku; Takahashi, Ishio; Shimizu, Michio; Tayama, Yoshinobu. *Journal of Materials Science (United Kingdom);* 26: No. 9, 2379-2388 (1 May 1991).

The corrosion of the coating-layers of silicon carbide (SiC) by carbon monoxide (CO) was observed in irradiated Triso-coated uranium dioxide particles, used in high-temperature gas-cooled reactors, by optical microscopy and electron probe micro-analysis. The mechanical failure of the coating-layer of inner dense pyrolytic carbon (IPyC) was often observed beside the area of the SiC corrosion. The grain boundaries of the SiC seemed to be selectively corroded during the early stages of corrosion. Silicon dioxide, or more stable (Si, Ce, Ba) oxide, was accumulated at the buffer-IPyC and IPyC-SiC interfaces on the cold side of the particles and the formation of (Pd, Rh, Ru, Tc, Mo) silicides was observed in the fuel kernels, which probably resulted from the vapour transport of silicon monoxide from the corroded areas. (author).

239
Corrosion. Taylor, D.F. (to General Electric Co., Schenectady, NY (USA)). USA Patent 4,986,957/A/. 22 Jan 1991. Filed date 25 May 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

A corrosion resistant nuclear fuel element. It comprises: an elongated cladding container formed from a zirconium alloy tube consisting essentially of by weight percent 0.5 to 2.0 percent tin, 0.24 to 0.40 percent of a solute composed of copper, nickel and iron, and the balance zirconium; and a central core of a body of nuclear fuel material selected from the group consisting of compounds of uranium, plutonium, thorium and mixtures thereof disposed in an d partially filling the container so as to leave a gap between the container and the core and an internal cavity at one end of the container an enclosure integrally secured and sealed at each end of the container and a nuclear fuel material retaining means positioned in the cavity.

FUEL ELEMENTS

240
(ANL/RERTR/TM–14)
The Whole-Core LEU U_3Si_2-Al Fuel Demonstration in the 30-MW Oak Ridge Research Reactor. Bretscher, M.M.; Snelgrove, J.L. (Argonne National Lab., IL (United States)). Jul 1991. Contract W-31109-ENG-38. 311p. Distribution: UC-520. OSTI; NTIS; INIS; GPO Dep. Order Number DE92000351.

The ORR Whole-Core LEU Fuel Demonstration, conducted as part of the US Reduced Enrichment Research and Test Reactor Program, has been successfully completed. Using commercially-fabricated U_3Si_2-Al 20%-enriched fuel elements (4.8 g U/cc) and fuel followers (3.5 g U/cc), the 30-MW Oak Ridge Research Reactor was safely converted from an all-HEU core, through a series of HEU/LEU mixed transition cores, to an all-LEU core. There were no fuel element failures and average discharge burnups were measured to be as high as 50% for the standard elements and 75% for the fuel followers. Experimental results for burnup-dependent critical configurations, cycle-averaged fuel element powers, and fuel-element-averaged ^{235}U burnups, validated predictions based on three-dimensional depletion calculations. Calculated values for plutonium production and isotopic mass ratios as functions of ^{235}U burnup support the corresponding measured quantities. In general, calculations for ^{60}Co and ^{198}Au reaction rate distributions, differential and integral control rod worths, prompt neutron decay constants, and isothermal temperature coefficients were found to agree with corresponding measured values. Experimentally determined critical configurations for fresh HEU and LEU cores radially reflected with water and with beryllium are well-predicted by both Monte Carlo and diffusion calculations. 44 refs., 57 figs., 45 tabs.

241
(ECN-R–91-002)
Some fundamental aspects of thermal conductivity of heterogeneously composed SPHERE-PAC fuel. Linde, A. van der. (Netherlands Energy Research Foundation, Petten (Netherlands)). Mar 1991. 63p. Photo-copies available from Library KNAW; P.O.Box 41950, 1009 DD Amsterdam, The Netherlands.

The standard fuel in LWR's consists of UO_2 pellets. The main draw-back of UO_2 is its low thermal conductivity. A 3-fraction mixture of large UO_2 spheres and medium and small spheres of a material with a high thermal conductivity, vibrationally compacted in a cladding tube, should give a sphere-pac fuel column with a higher thermal conductivity than pure UO_2 fuel. Evaluation of some fundamental aspects of the thermal conductivity of such a fuel column showed that sintering, i.e. necking, together of the medium and small spheres is required to achieve that higher overall thermal conductivity. A sphere-pac column consisting of 52 vol.percent large UO_2 spheres, 18 vol.percent medium SiC spheres, 18 vol.percent small SiC spheres and 12 vol.percent helium at 0.4 MPa between the spheres appeared to have potential as low as a new LWR fuel with a higher thermal conductivity. With 20 percent necking of the SiC spheres in such a UO_2-SiC sphere-pac fuel-column, the temperature in the centre is about 800 deg C at 30 kW/m, while in a standard UO_2 pellet stack it is then about 1100 deg. C. However, irradiation effects in the SiC spheres and fission gas contamination of the helium filling gas in the fuel rod cause a substantial decrease of the thermal conduc-tivity of the as-fabricated UO_2-SiC sphere-pac fuel column. The final conclusion is therefore that pelletizing a mixture of UO_2 and SiC spheres is probably a better way to achieve a high conductivity fuel for LWR's than vibratory compaction. (author). 18 refs.; 26 figs.; 9 tabs.

242
WIDAFELS flexible automation systems. Shende, P.S.; Chander, K.P.; Ramadas, P. pp. 395-399 of Symposium on advanced remote handling systems and automation in nuclear installations (ROSYMP-90) (held at Bombay during March 21-23, 1990): Proceedings. Bombay (India); Bhabha Atomic Research Centre (1990). 427p. (CONF-9003277–).

From Symposium on advanced remote handling systems and automation in nuclear installations; Bombay (India) (21-23 Mar 1990).

After discussing the various aspects of automation, some typical examples of various levels of automation are given. One of the examples is of automated production line for ceramic fuel pellets. (M.G.B.).

243
Quality assurance in a nuclear tubing mill. Badri Narayan, J. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 125-131 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

The Specialty Metals Plant Quality Assurance system and the quality organization are described in general. Hierarchy of the system documents and the dual functions of the Plant Product Assurance Department are described. The necessary modifications to the plant procedures to meet the international QA requirements are discussed. (orig.).

244
Fabrication of UO_2 fuels with nondestructive assay and rod scanner technology for production and final quality control. Dams, W.; Baumann, R.; Hanel, I.; Happ, B.; Heins, L. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 171-178 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

The production of nuclear fuels requires a continuous in-process inspection which includes the control of equipment and process parameters as well as the quality control of the material. All components and intermediate products belonging to fuel fabrication, or being produced throughout the process, are examined by physical and chemical tests. The supervision of important process and quality parameters was improved by new techniques and instrumentations of nondestructive in-line and laboratory assays. For example the ^{235}U enrichment is supervised by the UF_6-incoming control up to a continuously working in-line instrumentation, which measures the ^{215}U concentration in the feed hoppers before pressing. For final control of the fuel rods, an active rod scanner, which examines the rod's structure and composition, the ^{235}U enrichment, and pellet mix-up, is used. (orig.).

245
Statistical analysis of QC data and estimation of fuel rod behaviour. Heins, L.; Gross, H.; Nissen, K.; Wunderlich, F. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 287-295 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

The behaviour of fuel rods while in reactor is influenced by many parameters. As far as fabrication is concerned, fuel pellet diameter and density, an inner cladding diameter are important examples. Statistical analyses of quality control data show a scatter of these parameters within the specified tolerances. At present it is common practice to use a combination of superimposed unfavorable tolerance limits (worst case dataset) in fuel rod design calculations. Distributions are not considered. The results obtained in this way are very conservative but the degree of conservatism is difficult to quantify. Probabilistic calculations based on distribution allow the replacement of the worst case dataset by a dataset leading to results with known, defined conservatism. This is achieved by response surface methods and Monte Carlo calculations on the basis of statistical distributions of the important input parameters. The procedure is illustrated by means of two examples. (orig.).

246
Determination of the distribution of the largest Pu-rich particles in the pellets of a batch. Vollath, D. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 296-299 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

It is shown that by means of advanced methods of statistics estimates can be made on the sizes of the largest plutonium rich particles in the pellets of a batch. The prerequisite is that the data obtained from a cutting plane, e.g. by autoradiography and image analysis or with the help of a microprobe, can be converted into three-dimensional data. When it is possible to fit these data to a distribution function capable of extrapolation, the distribution function of the largest particles can be calculated. Obviously, this calls for a *homogeneous distribution of plutonium* in the different pellets which must be verified. The method used can be applied to other similar problems as well such as the sizes of the gadolinium rich zones. (orig.).

247
^{252}Cf-source-driven neutron noise measurements of subcriticality for an annular tank containing aqueous Pu-U nitrate. Mihalczo, J.T.; Blakeman, E.D.; Ragan, G.E.; Kryter, R.C.; Seino, H.; Robinson, R.C. *Nuclear Technology (United States);* 94: No. 3, 336-360 (Jun 1991).

A series of subcritical experiments in unreflected annular geometry was performed with an aqueous Pu-U nitrate containing 173 and 262 g/ℓ of plutonium and uranium, respectively. The plutonium contained 91.1 wt% ^{239}Pu, while the depleted uranium contained 0.57 wt% ^{235}U. In these experiments, the height of the solution in the annulus was varied from 17.8 to 84.2 cm. The annulus had an inner diameter of 25.4 cm, an outer diameter of 53.34 cm, and a 0.08-cm-thick wall of Type 304L stainless steel. In this paper, measurements using the ^{252}Cf-source-driven neutron noise analysis method were interpreted to obtain the subcritical neutron multiplication factors. The data accumulated in the experiment, which is the first test of this method in annular geometry, are summarized, and the analysis of these data is presented. The results and conclusions of these experiments are as follows: the capability to measure the sub-criticality for a multiplying system of annular geometry to a k as low as 0.70 was demonstrated; the criteria developed in previous experiments for choosing source-detector system configurations for which the data can be interpreted using a modified point kinetics were also satisfactory for this experiment; the measurement times for this geometry were not significantly different from those used for cylindrical geometry and were sufficiently short to allow practical measurements; the reactivities obtained from break frequency noise analysis measurements agreed with those obtained from the ratios of spectral densities within the experimental uncertainties; the applicability of the method and an understanding of the theory of the measurement method for plutonium solution systems were demonstrated; and the calculated neutron multiplication factors agreed with the experimental values of k to within ~0.03.

248
Eddy current examination technique during manufacturing of Zircaloy-4 fuel cladding tubes. Senevat, J.; Mainy, P. *Journal of Nuclear Materials (Journal des Materiaux Nucleaires) (Netherlands);* 178: No. 2/3, 315-320 (Feb 1991). (CONF-900625–).

From Characterization and quality control of nuclear fuels; Karlsruhe (Germany) (19-21 Jun 1990).

After reviewing the examination methods traditionally used for detecting localized defects (ultrasonic testing and visual inspection) an original examination method based on the eddy current technique is presented. Specific examples of application are then used to illustrate the ability of this technique to detect defects which are hard to reveal with other methods. The interlocking of the eddy current and ultrasonic techniques is illustrated: the capabilities of each method are developed and the defect characteristics corresponding to the indications are described. Finally, the advantages of combining the visual, ultrasonic and eddy current examination techniques are analyzed and the paper concludes by referring to the quality improvement obtained for the same high level of manufacturing productivity. (orig.).

249
Nuclear fuel assembly spacer and loop spring with enhanced flexibility. Johansson, E.B. (to General Electric Co., San Jose, CA (USA)). USA Patent 5,002,726/A/. 26 Mar 1991. Filed date 27 Dec 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes improvement in a loop spring for maintaining fuel rods in spaced apart relation in a fuel bundle space. The spacer including; first and second overlying spring supporting elements for supporting a spring. The spring supporting elements disposing the spring on opposite sides to and towards a ferrule for containing fuel rods. The spring elements of the type formed from a continuous elongated loop of spring material. The spring element in the unstressed state having the loop defining an upper C-shaped end in the loop for surrounding and being supported from the first spring support member from the spacer; a lower C-shaped end in the loop for surrounding and being supported from a second and underlying spacer member; paired spring leg members on either side of the loop for closing the loop between the C-shaped ends; the paired spring leg member further defining medially thereof a convex and outwardly disposed arch shaped rod contacting portion, the rod contacting portions on each of the paired spring leg members being medially located in the spring leg members between the upper and lower C-shaped section.

250
Pressure pulse method and system for removing debris from nuclear fuel assemblies. Cadwell, D.J.; Franklin, R.D. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). USA Patent 5,002,079/A/. 26 Mar 1991. Filed date 15 Dec 1988. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes a system for removing debris from a nuclear fuel assembly of the type having a plurality of fuel rods, each of which is engaged by spring retaining means within a grid. It comprises means for securing the fuel assembly within a pool of water; means for isolating the water surrounding the fuel assembly in the securing means, and at least one pressure pulse source in communication with the water isolated within the isolation means for discharging a series of pulses of pressurized gas within the isolated water that create shock waves for exerting momentary forces on the fuel rods sufficient to dislodge debris but insufficient to cause any of the fuel rods to momentarily disengage the spring retaining means.

251
Nuclear fuel assembly with expandable top nozzle subassembly. Sparrow, J.A.; Lee, Y.C. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). USA Patent 4,986,959/A/. 22 Jan 1991. Filed date 17 May 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes an expandable top nozzle subassembly for a nuclear fuel assembly. It comprises: an upper structure; a lower adapter plate; interengagable means; and resiliently-yieldable biasing devices.

252
Nuclear fuel pellet surface defect inspection apparatus. Ahmed, H.J. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). USA Patent 4,978,495/A/. 18 Dec 1990. Filed date 26 Oct 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This paper describes an apparatus for inspecting nuclear fuel pellets for surface defects and having an inspection chamber, a pellet guide chute assembly extending through the inspection chamber. It comprises: a support substrate composed of a pair of elongated wall sections each having a pair of opposite spaced longitudinal edges. The wall sections being rigidly attached together along adjacent ones of the longitudinal edges to form a corner and being angularly displaced from one another at remote ones of the longitudinal edges to have a generally V-shaped configuration in cross section; a pair of elongated plates respectively disposed on upper adjacent sides of the support substrate wall sections; and means for adjustably attaching the plates to the wall sections for sliding movement thereon toward and away from the adjacent longitudinal edges thereof to locate the plates at desired stationary positions thereon for supporting nuclear fuel pellets of different diameters between and on adjacent longitudinal portions of the plates located in the inspection chamber of the apparatus.

253
Nuclear fuel element, and method of forming same. Taylor, I.N. Jr.; Magee, P.M. (to General Electric Co., San Jose, CA (USA)). USA Patent 4,971,753/A/. 20 Nov 1990. Filed date 23 Jun 1989. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes improvement to the method of inhibiting an interaction between a metal alloy fissionable fuel for a nuclear reactor and a stainless steel container for the fuel which results in low temperature melting eutectic reaction products of components from the metal alloy fuel and stainless steel. The improvement comprises providing an expandable body of at least one alloying metal for the metal alloy fissionable fuel selected from the group consisting of zirconium, titanium, niobium and molybdenum at least about 2 mils thick interposed into a space between a metal alloy fuel. The fuel consists of metallic uranium and plutonium and the stainless steel container housing the metal alloy fuel therein.

254
Nuclear fuel bodies and the production thereof. Wood, G.A. UK Patent 2235816/A/. 13 Mar 1991. 4p. Available from The Patent Office, Sales Branch, Unit 6, Nine Mile Point, Cwmfelinfach, Cross Keys, Newport, NP1 7HZ.

A nuclear fuel body comprises doped uranium dioxide grains having kernels of undoped uranium dioxide. The body is produced by mixing single crystal seeds of uranium dioxide with doped uranium dioxide granules. The mixture is compacted and subsequently sintered. Gadolinia is a preferred dopant for the uranium dioxide. (Author).

255
Method for non-destructive measurement of heat affected zone of identification code on nuclear fuel rod. Goldenfield, M.P.; Lambert, D.V.; Shannon, R.E. UK Patent 2234624/A/. 6 Feb 1991. 18p. Available from The Patent Office, Sales Branch, Unit 6, Nine Mile Point, Cwmfelinfach, Cross Keys, Newport, NP1 7HZ.

In a method for the nondestructive measurement of the depth of the heat affected zone underneath a laser generated identification code on a nuclear fuel rod tube, the change in impedance of an electromagnetic foil produced by a portion of the tube is measured before and after the etching of a bar code is performed by use of laser power. The impedance change in the coil produced by the bar code-bearing tube portion is compared with the impedance change produced in the coil by the same tube portion before the bar code is generated thereon to determine whether a maximum allowed depth of the heat affected zone of the tube portion has been exceeded. (author).

CONTROL SYSTEMS

256
(ANL/CP-72713)
The use of modern control methods to improve liquid-metal reactor safety and availability. Vilim, R.B. (Argonne National Lab., IL (United States)). [1991]. Contract W-31109-ENG-38. 7p. (CONF-911014-2). OSTI; NTIS; INIS; GPO Dep. Order Number DE91018616.

From International conference on dynamics and control in nuclear power stations; London (United Kingdom) (22-24 Oct 1991).

The development of metal fuel for fast reactors has created a new opportunity for innovation in plant control. By exploiting the superior neutronics properties of metal fuel, the reactivity burnup swing can be made zero, obviating the need for control rod reactivity addition over life. This result can be used to simplify elements of the plant control and protection system, the goal being improved plant safety and availability. These simplifications, however depend on the development of methods for improved reactor inlet temperature control and reactivity feedback monitoring. The use of modern control theory to develop these methods is described. 10 refs., 4 figs.

257
(ANL/CP-73862)
Detection and diagnosis of abnormal transients in nuclear power plants. Lee, J.C.; Rank, P.J.; Hawkes, E.; Wehe, D.K.; Reifman, J. (Argonne National Lab., IL (United States)). [1991]. Contract W-31109-ENG-38. 6p. (CONF-910983-2). OSTI; NTIS; INIS; GPO Dep. Order Number DE91018643.

From ENFIR Brazilian meeting on reactor physics and thermal hydraulics; Sao Paulo (Brazil) (17-20 Sep 1991).

This document describes a simulation-based algorithm that combines fuzzy logic with macroscopic conservation equations to diagnose multiple-failure events subject to uncertainties in transient data. Clusters of single-failure data points of similar characteristics are obtained through a pattern recognition algorithm and the cluster centers are combined in the space of macroscopic inventory derivatives to generate multiple-failure cluster centers. A fuzzy membership function is used to represent the likelihood of a data point belonging to a cluster, and failure estimates are obtained through solution of a fuzzy matrix equation. The algorithm has been successful in detecting simulated malfunctions in the pressurizer of a pressurized water reactor. 11 refs., 9 figs., 1 tab.

258
(ECN-RX–91-025)
Failure detection by adaptive lattice modelling using Kalman filtering methodology : application to NPP. Ciftcioglu, O.; Turkcan, E. (Netherlands Energy Research Foundation, Petten (Netherlands)). Mar 1991. 16p. (CONF-910535–). OSTI; NTIS (US Sales Only); INIS. Order Number DE92603277.

From 6. specialists meeting on reactor noise (SMORN); Gatlinburg, TN (United States) (19-24 May 1991).

Detection of failure in the operational status of a NPP is described. The method uses lattice form of the signal modelling established by means of Kalman filtering methodology. In this approach each lattice parameter is considered to be a state and the minimum variance estimate of the states is performed adaptively by optimal parameter estimation together with fast convergence and favourable statistical properties. In particular, the state covariance is also the covariance of the error committed by that estimate of the state value and the Mahalanobis distance formed for pattern comparison takes x^2 distribution for normally distributed signals. The failure detection is performed after a decision making process by probabilistic assessments based on the statistical information provided. The failure detection system is implemented in multi-channel signal environment of Borssele NPP and its favourable features are demonstrated. (author). 29 refs.; 7 figs.

259
(SU-DCE-RR–419)
Fault detection using optimal control techniques. Jalel, N.A.; Nicholson, H. (Sheffield Univ. (United Kingdom)). Nov 1990. 24p. Available from British Library Document Supply Centre, Boston Spa, Wetherby, West Yorks. LS23 7BQ.

The possibility of using optimal control techniques for fault diagnosis is investigated in this paper. To achieve this, two techniques will be used. In the first, the performance index will be used to identify the abnormal behaviour in any subsystem of the Loss Of Fluid Test (LOFT) reactor, a small scale pressurised water reactor, and in the other, the faulty controller of the LOFT reactor will be identified using the Pontryagin maximum principle. (author).

260
(ZJE–285)
Advanced version of LKP-M step drive. Haniger, L.; Elger, R.; Kocandrle, L.; Zdebor, J. (Skoda, Plzen (Czechoslovakia)). 1990. 12p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605440.

The control rod drive for WWER-1000 type reactors is described in detail. The total working length of the draw bar is 3500 mm. The lower section is equipped with a spring type shock absorber. Two types of latches, viz. fixing latches and drawing latches, are used; their opening and closing are controlled through conical bushings connected by concentric tubes to the moving poles of the electromagnets. Inductance coils are used for the cluster position indication. The power package is fed from an independent 3x220 V, 50 Hz system, whereas a 110 V accumulator serves the standby feeding. The accuracy of the position indication is 1%. The functions of the power control and of the position indicator are described in detail. (M.D.).

261
C and I power supply for research reactors (Paper No. CP 33). Joseph, Jose; Patil, R.K. Bombay (India); Bhabha Atomic Research Centre (1990). 11p. (CONF-9002196–).

From National symposium on DC and RF power sources in research industry; Bombay (India) (26-28 Feb 1990).

The safety and availability of a nuclear reactor depends greatly on the quality and reliability of control and instrumentation (C and I) power supply. It may not be practical to incorporate fail-safe philosophy in all safety systems of a reactor due to fact that one has to pay huge penalty for spurious actuation of certain systems. A reliable power supply system can solve this problem. Availability of the status of reactor and various systems is essential for the safe operation of the plant. In small size reactors the C and I is powered from normal power supply. It is not practical to provide battery power backup for the whole C and I due to charging and maintenance problems associated with large batteries. This is more so since the reactor is not operated on a continuous basis. But it becomes mandatory to provide some amount of battery power atleast to indicate certain vital information like reactor status etc. in case of failure of normal power supply. This paper describes various aspects of C and I power supply for the research reactors DHRUVA, PURNIMA and KAMINI and highlights their evolution in order to ensure safe and reliable operation of the reactor. (author). 6 refs.

262
Nuclear reactor control and protection instrumentation in China. Binsen, J. *Nuclear Engineer (Institution of Nuclear Engineers) (United Kingdom);* 32: No. 1, 16-21 (Jan-Feb 1991).

This paper introduces the general situation of nuclear reactor instrumentation in China in the past, nowadays and in future. In China the Ministry of Nuclear Industry undertakes all of the tasks for nuclear reactor design, research, manufacture and operation. The design and manufacture of nuclear reactor control and protection instruments are completed by research institutes and factories which belong to the Ministry of Nuclear Industry. At the beginning of 1956 China imported the first test heavy water nuclear reactor from USSR. We started to learn nuclear reactor technology and control and protection. Research and development on reactor control and protection instrumentation has continued. China started to establish national standards of reactor instrumentation. Control and protection systems of Qin Shan nuclear power plant and the first low temperature thermo supply nuclear reactor have been designed, and are being constructed to national standards, which are similar to international standards. We type tested the reactor control and protection system of the nuclear power plant putting these instruments into circumstances which simulated the conditions of a nuclear power plant for a long time to examine their qualities and characteristics. Testing demonstrated that quality of these instruments is high. With the need for systems for the Qin Shaw nuclear powerplant, research and development of new reactor control and protection instrumentation also began. The main purpose of development is to use microprocessors and optical fibre technology in reactor control and protection instrumentation. Testing and maintenance ability of the integrated control and protection system will be higher than before. The availability and reliability of nuclear reactors will be increased by using these new instruments. (Author).

263
Dynamic oscillations predicted by computer studies. Butts, M.M.; Smith, H.S. *IEEE Computer Applications and Power (United States);* 4: No. 1, 47-51 (Jan 1991).

During the latter part of 1988, a study was begun to review the dynamic stability performance of a power company's plant. The scope of the study was to identify any operating conditions that might contribute to system oscillations and to examine alternative solutions that would control these oscillations. The study was performed in several phases. This paper discusses the study process, utilizing two different software packages for the analysis: Dynamic stability studies using time-domain software and Eigenvalue analysis using frequency-domain software.

264
The computer-based process information system for the 5 MW THR. Zhang Liangju; Zhang Youhua; Liu Xu; An Zhencai; Li Baoxiang. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 81-84 (Oct 1990). (In Chinese).

The computer-based process information system has effectively improved the interface between operation person and the reactor, and has been successfully used in reactor operation environment. This article presents the design strategy, functions realized in the system and some advanced techniques used in the system construction and software development.

265
Safety analysis and evaluation of hydraulic control rod drive system used for 5 MW THR. Hu Yuedong; Wu Yuanqiang; Liu Chengying; Cheng Yunsheng; An Jian; Dong Yonggui. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 77-80 (Oct 1990). (In Chinese).

A new type of hydraulic control rod drive system (HCRDS) was used for 5 MW THR. Because its design is beased on passive system and the drive and guide of control rod are integrated, this system possesses reliable inherent safety feature and assures reactor safety under any accident conditions.

266
Design and experimental research of hydraulic control rod drive system for 5 MW THR. Wu Yuanqiang; Hu Yuedong; Cheng Yunsheng; Yang Nianzu; Liu Chengying; Zhang Fulu. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 73-76 (Oct 1990). (In Chinese).

Hydraulic control rod drive system (HCRDS) is a new device of control rod drive which is different from usual electric-magnetic mechanic drive used for water power nuclear reactor. The coolant(water) is used as actuating medium, pumped by canned-pump, then injected into hydraulic step cylinders which are set in the reactor vessel. The outer tube of the step cylinder will be moving step by step by controling of the flow, driving the neutron absorber. 5MW THR is the first reactor in the world in which the hydraulic step drive system was used. Using of the drive system is for getting better safety, reliability and lower cost. In this paper, the design and experimental research of the system are introduced.

267
The control system overall design for the 5 MW THR. Guo Renjun; Feng Zhiyi; Yang Zijue; Li Baoxiang; Zhou Shixin. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 70-72, 89 (Oct 1990). (In Chinese).

The system provides the means of monitor,control and protection for 5 MW THR. It covers six subsystems, they are the nutron flux measurement system, the reactor protection system, the reactor control system, the alarm system, the on-line process computer system and the central control room. The control system overall design is based on the type and the use of 5 MW THR, as well as nuclear and thermal characteristics of the reactor. The design follows the relative code and standards for nuclear power plants safety design.

268
The visible study on the mass transfer model for 5 MW THR. Zha Meisheng; Nie Mengchen; Zhou Huizhong; Wang Liqun; Guo Weiping; Liu Zhiyong. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 65-69 (Oct 1990). (In Chinese).

A boron solution injection system is used as the standby liquid control system for 5MW THR. It is necessary for design of a nuclear reactor mixing with the coolant. This paper describes a visible study, which obviously displays the boron mixing with water in the main loop for differnt values of $(\rho_j\omega_j)/(\rho_p\omega_p)$.

269
A fault detection and diagnosis in a PWR steam generator. Park, Seung Yub. *Chongi Hakhoe Nonmunchi (Transactions of the Korean Institute of Electrical Engineers) (Korea, Republic of);* 40: No. 1, 120-127 (Jan 1991). (In Korean).

The purpose of this study is to develop a fault detection and diagnosis scheme that can monitor process fault and instrument fault of a steam generator. The suggested scheme consists of a Kalman filter and two bias estimators. Method of detecting process and instrument fault in a steam generator uses the mean test on the residual sequence of Kalman filter, designed for the unfailed system, to make a fault decision. Once a fault is detected, two bias estimators are driven to estimate the fault and to discriminate process fault and instrument fault. In case of process fault, the fault diagnosis of outlet temperature, feed-water heater and main steam control valve is considered. In instrument fault, the fault diagnosis of steam generator's three instruments is considered. Computer simulation tests show that on-line prompt fault detection and diagnosis can be performed very successfully.(Author).

270
A digital data acquisition approach for research reactors. Ficaro, E.P.; Wehe, D.K. *Nuclear Technology (United States);* 94: No. 2, 262-269 (May 1991).

This paper describes a digital data acquisition and display system for research reactor applications. Prototype and commercially available hardware is presented for the analog and digital signals. Particular problems encountered in actually implementing the system are highlighted. Three applications showing the advantages of the digital monitoring system are presented.

271
Power prediction in nuclear power plants using a back-propagation learning neural network. Roh, M.; Cheon, S.; Chang, S. *Nuclear Technology (United States);* 94: No. 2, 270-278 (May 1991).

This paper proposes an artificial neural network - a data processing system with a number of simple highly interconnected processing elements in an architecture inspired by the structure of the human brain for the prediction of thermal power in nuclear power plants (NPPs). The back-propagation network (BPN) algorithm is applied to develop models of signal processing. A number of case studies are performed with emphasis on the applicability of the network in a steady-state high power level. The studies reveal that the BON algorithm can precisely predict the thermal power of an NPP. It also shows that the defected signals resulting from instrumentation problems, even when the signals comprising various patterns are noisy or incomplete, can be properly handled.

272
Use of expert judgement in NUREG-1150. Ortiz, N.R.; Wheeler, T.A.; Breeding, R.J.; Hora, S.; Meyer, M.A.; Kenney, R.L. *Nuclear Engineering and Design (Netherlands);* 126: No. 3, 313-331 (May 1991).

The explicit expert judgment process used in NUREG-1150, 'Severe Accident Risks: An Assessment for Five US Nuclear Plants', is discussed in this paper. The main steps of the process are described, including selection of issues and experts, elicitation training, presentation of issues to the experts, preparation of issue analyses by the experts, discussion of issue analyses and elicitation, and recomposition and aggregation of results. To demonstrate the application of the expert judgment process to NUREG-1150, two issues are summarized: one from the accident frequency analysis, and one from the accident progression analysis. Recommendations and insights are provided to improve the use of explicit expert judgment in complex technical issues. (orig.).

273
Derivation of consistent reactivity worth and eigenvalue separation from space-dependent rod worths on the basis of modal approach. Hashimoto, K.; Ohsawa, T.; Miki, R.; Shibata, T. *Annals of Nuclear Energy (United Kingdom);* 18: No. 6, 317-325 (1991).

A method is proposed to infer the consistent reactivity worth and the λ-mode eigenvalue separation from space-dependent control rod worths in loosely-coupled reactors. Using the modal expansion approximation for a transient flux, we derive a two-mode version of the formula for an integral-count rod drop measurement. The formula is very simple and there is no need for a theoretical correction factor. The present formula is more general than the previous two-point formula, and is anticipated to be applicable even to large single-core reactors. The applicability of the formula is confined by a demonstration in the UTR-KINKI reactor, a light-water moderated and graphite-reflected reactor. (author).

274
Rule-based fuzzy logic controller for a PWR-type nuclear power plant. Akin, H.L.; Altin, V. *IEEE Transactions on Nuclear Science (Institute of Electrical and Electronics Engineers) (United States);* 38: No. 2, 883-890 (Apr 1991).

This paper reports on a rule-based fuzzy logic regulator developed for a PWR-type nuclear power plant based on knowledge acquisition through numerical simulations and making use of a validated mathematical model of the H.B. Robinson power plant. Production rules were used for knowledge representation and fuzzy sets were implemented using broken line. Due to the nature of the rules, forward chaining was selected as the inferencing mechanism. Using an error criterion, the gain and sampling interval values were adjusted. The behavior of this rule-based controller was investigated under both normal and noisy operating conditions and in the presence of drift in process variables. It was observed that there was negligible degradation in the performance of the controller in the presence of noise and drift in process variables.

275
Demonstration of feedback using the MIT-SNL minimum time control laws for the rapid maneuvering of reactor power. Bernard, J.A. *IEEE Transactions on Nuclear Science (Institute of Electrical and Electronics Engineers) (United States);* 38: No. 2, 838-844 (Apr 1991).

This paper reports on experiments in which the efficacy of combining proportional-integral-derivative feedback methodologies with the MIT-SNL Period-Generated Minimum Time Control Laws that were demonstrated for the safe maneuvering of reactor neutronic power on periods as short as 1.0 s under conditions of closed-loop digital control. It is shown that proportional feedback cannot compensate for errors in the measurement of reactivity and that the addition of integral action corrects for this deficiency. The need for controllers of safety-constrained systems to be capable of not only compensating for perturbations but also of diagnosing them as to cause is discussed.

276
Process hypercube comparison for signal validation. Holbert, K.E. *IEEE Transactions on Nuclear Science (Institute of Electrical and Electronics Engineers) (United States);* 38: No. 2, 803-811 (Apr 1991).

The optimal control and safe operation of a nuclear power plant requires reliable information concerning the state of the process. Signal validation is the detection, isolation, and characterization of faulty signals. Properly validated process signals are beneficial from the standpoint of increased plant availability and reliability of operator actions. This paper reports on a signal validation technique utilizing a process hypercube comparison (PHC) originated during this research. The hypercube is merely a multidimensional joint histogram of the process conditions. The hypercube is created off-line during a learning phase using operational plant data. In the event that a newly observed plant state does not match with those in the learned hypercube, the PHV algorithm performs signal validation by progressively hypothesizing that one or more signals is in error. This assumption is then either substantiated or denied. In the case where many signals are found to be in error, a conclusion that the process conditions are abnormal is reached. The global data base contained within the hypercube provides a best estimate of the process conditions in the event a signal is deemed failed. The hypercube signal validation methodology was tested using operational data from a commercial pressurized water reactor (PWR) and the Experimental Breeder Reactor II (EBR-II). This research was part of a larger project aimed at the development of a comprehensive signal validation software system for application to nuclear power plants.

277
Hydraulic lock for displacer rod drive mechanism (DRDM) and method of operation. Rinker, E.D. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). USA Patent 4,978,484/A/. 18 Dec 1990. Filed date 1 Apr 1988. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This paper describes a drive rod latch in combination with a nuclear reactor having a drive rod disposed in a rod housing characterized in that the drive rod has one end selectively exposed to a first, relatively low pressure zone of the reactor and another end thereof in communication with a second, relatively high pressure zone of the reactor. The drive rod further having disposed on an end thereof a valve member and the rod housing having disposed thereon a corresponding valve seat, and a control valve for selectively establishing communication between the housing and the first zone of the reactor whereby a pressure differential is created across the piston. The pressure differential being sufficient to seat the valve member against the valve seat to thereby establish a pressure boundary.

278
Control system for a nuclear steam power plant. Regan, J.A.; Estrada, H. Jr. (to MPR, Inc., Washington, DC (USA)). USA Patent 4,975,238/A/. 4 Dec 1990. Filed date 1 Sep 1988. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes a control system for a nuclear steam power plant having a nuclear reactor, a once through steam generator, means circulating coolant between the nuclear reactor and the once through steam generator. It includes a hot coolant pipe, a cold coolant pipe and coolant pump means, means supplying feedwater to the once through steam generator including feedwater flow regulating means for controlling feedwater supplied to the once through steam generator, a turbine driven by steam generated by the once through steam generator and means supplying steam from the once through steam generator to the turbine including governor valve means for controlling the amount of steam supplied to the turbine. It comprises: a reactor control subsystem; a feedwater control subsystem; and a turbine steam demand control subsystem.

279
Wear-reduction-sleeve for thimbles. Haslinger, K.H.; Porter, D.S.; Martin, M.L.; Higgins, W.H. (to Combustion Engineering, Inc., Windsor, CT (USA)). USA Patent 4,975,241/A/. 4 Dec 1990. Filed date 29 Jul 1988. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This patent describes a nuclear reactor having a core support plate with an annulus therein, a fuel assembly distribution-plate with an opening therein and spaced from the core support plate, an in-core-instrumentation thimble, a thimble guide tube, and a wear-reduction-sleeve for the in-core-instrumentation thimble, to curtail flow induced excitation in a region of thimble length which extends through the annulus and upwardly to and through the opening. It comprises: means for surrounding the thimble in the portion of its region of length in the annulus and substantially filling the annulus to increase support contact surface area; and means for collecting a major portion of thimble flow and diverting it radially away from the wear-reduction-sleeve and the opening.

280
System for removing and installing a control rod drive. Larson, R.C.; Spencer, K.R.; Alercia, D.P.; Kweck, H.; Litka, T.J.; Ford, G.J. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). USA Patent 4,973,443/A/. 27 Nov 1990. Filed date 26 Oct 1988. vp. Patent and Trademark Office, Box 9, Washington, DC 20232 (USA).

This paper discusses a system for removing and installing an elongated control rod drive from a drive housing mounted in the vessel of a boiling water reactor. The reactor is of the type having an undervessel cavity with a pair of service rails one of which includes a tapered surface along its upper edge.

ENVIRONMENTAL ASPECTS

281
(AERE-M-3806)
Harwell Laboratory: radioactive discharges and environmental monitoring. (AEA Technology, Harwell (United Kingdom)). Jun 1991. 44p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92602981.

The United Kingdom Atomic Energy Authority's Harwell Laboratory is situated some 5 miles west of Didcot in Oxfordshire. The AEA Businesses located at the Laboratory carry out a wide range of nuclear and non-nuclear investigations for customers in the UK and overseas. This report provides information on disposal of radioactive waste from Harwell and on associated environmental monitoring for the calendar year 1990. In most instances information has also been provided on measurements made in previous years to show the 1990 results in their historical perspective. The National Radiological Protection Board, (NRPB) currently recommends a primary dose limit for individual members of the public of 1 mSv per year from sources other than natural background radiation, and that operations of a single site such as Harwell should give rise to a dose of no more than 0.5 mSv per year. No member of the public is judged to have exceeded this dose as a consequence of discharges made in 1990. Throughout this report, discharge measurements have been assessed against derived limits corresponding to a dose of 0.5 mSv, consistent with NRPB advice. At the end of March 1990, the Harwell materials testing reactors, DIDO and PLUTO, were shut down for the final time after over 34 years' operation. Later in the year the GLEEP reactor was closed down after 43 years of continuous operation, the longest serving reactor in the world. (author).

282
(NUREG-1437-Vol.1)
Generic Environmental Impact Statement for license renewal of nuclear plants. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution). Aug 1991. 553p. OSTI; NTIS; INIS; GPO.

The Nuclear Regulatory Commission (NRC) anticipates that it will receive applications for renewal of the operating licenses of a significant portion of existing nuclear power plants. This Generic Environmental Impact Statement (GEIS) examines the possible environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants under the proposed 10 CFR Part 54. The GEIS, to the extent possible, establishes the bounds and significance of these potential impacts. The analyses in the GEIS encompass all operating light-water power reactors. For each type of environmental impact the GEIS attempts to establish generic findings covering as many plants as possible. While plant and site-specific information is used in developing the generic findings, the NRC does not intend for the GEIS to be a compilation of individual plant environmental impact statements. This GEIS has three principal objectives: (1) to provide an understanding of the types and severity of environmental impacts that may occur as a result of license renewal of nuclear power plants under 10 CFR Part 54, (2) to identify and assess those impacts that are expected to be generic to license renewal, and (3) to support a rulemaking (10 CFR Part 51) to define the number and scope of issues that need to be addressed by the applicants in plant-by-plant license renewal proceedings. To accomplish these objectives, the GEIS makes maximum use of environmental and safety documentation from original licensing proceedings and information from state and federal regulatory agencies, the nuclear utility industry, the open literature, and professional contacts. 676 refs., 26 figs., 117 tabs.

283
(NUREG–1437-Vol.2)
Generic Environmental Impact Statement for license renewal of nuclear plants. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution). Aug 1991. 597p. OSTI; NTIS; INIS; GPO.

The Nuclear Regulatory Commission (NRC) anticipates that it will receive applications for renewal of the operating licenses of a significant portion of existing nuclear power plants. This Generic Environmental Impact Statement (GEIS) examines the possible environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants under the proposed 10 CFR Part 54. The GEIS, to the extent possible, establishes the bounds and significance of these potential impacts. The analyses in the GEIS encompass all operating light-water power reactors. For each type of environmental impact the GEIS attempts to establish generic findings covering as many plants as possible. While plant and site-specific information is used in developing the generic findings, the NRC does not intend for the GEIS to be a compilation of individual plant environmental impact statements. This GEIS has three principal objectives: (1) to provide an understanding of the types and severity of environmental impacts that may occur as a result of license renewal of nuclear power plants under 10 CFR Part 54; (2) to identify and assess those impacts that are expected to be generic to license renewal, and (3) to support a rulemaking (10 CFR Part 51) to define the number and scope of issues that need to be addressed by the applicants in plant-by-plant license renewal proceedings. To accomplish these objectives, the GEIS makes maximum use of environmental and safety documentation from original licensing proceedings and information from state and federal regulatory agencies, the nuclear utility industry, the open literature, and professional contacts. This document, Volume 2, provides the appendices to this report. 31 refs., 58 figs., 154 tabs.

284
(NUREG/CR–5634)
Identification and assessment of containment and release management strategies for a BWR Mark I containment. Lin, C.C.; Lehner, J.R. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research; Brookhaven National Lab., Upton, NY (United States)). Sep 1991. Contract AC02-76CH00016. 208p. (BNL-NUREG–52259). OSTI; NTIS; INIS; GPO.

This report identifies and assesses accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark 1 type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark 1 containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. A safety objective tree is developed which provides the connection between the safety objectives, the safety functions, the challenges, and the strategies. The strategies were assessed by applying them to certain severe accident sequence categories which have one or more of the following characteristics: have high probability of core damage or high consequences, lead to a number of challenges, and involve the failure of multiple systems. 59 refs., 55 figs., 27 tabs.

285
(RFP-Trans–503)
Process and device for cleaning of water-containing machine oil from nuclear power plants. Kupfer, K. (Rockwell International Corp., Golden, CO (United States). Rocky Flats Plant). Oct 1989. Contract AC34-90DP62349. 13p. Translation of German Patent Application No. 3,600,358 A1, 16 July 1987. OSTI; NTIS; INIS; GPO Dep. Order Number DE91018406.

The radioactive contaminants produced in water-containing machine oil in plants operated with nuclear power are removed from the contaminated machine oil by adsorption on water-adsorbing material. This document discusses this process. 3 figs., 1 tab.

286
(UCRL-CR–106554)
Structural concepts and details for seismic design. (Lawrence Livermore National Lab., CA (United States); EQE International (United States). Engineering and Design Div.). Sep 1991. Contract W-7405-ENG-48. 260p. OSTI; NTIS; GPO Dep. Order Number DE92001892.

This manual discusses building and building component behavior during earthquakes, and provides suggested details for seismic resistance which have shown by experience to provide adequate performance during earthquakes. Special design and construction practices are also described which, although they might be common in some high-seismic regions, may not be common in low and moderate seismic-hazard regions of the United States. Special attention is given to describing the level of detailing appropriate for each seismic region. The UBC seismic criteria for all seismic zones is carefully examined, and many examples of connection details are given. The general scope of discussion is limited to materials and construction types common to Department of Energy (DOE) sites. Although the manual is primarily written for professional engineers engaged in performing seismic-resistant design for DOE facilities, the first two chapters, plus the introductory sections of succeeding chapters, contain descriptions which are also directed toward project engineers who authorize, review, or supervise the design and construction of DOE facilities. 88 refs., 188 figs.

287
The siting of low temperature district heating reactor by the environmental impact of 5 MW THR. Liu Yuanzhong; Fang Dong. *Nuclear Power Engineering (Hedongli Gongcheng) (China);* 11: No. 5, 21-23, 44 (Oct 1990). (In Chinese).

Based on the Analyses of envirommental impact from radioactive releace during routine operation and accident for 5 MW THR, it is pointed out that district heating reactor (DHR) can be built in suburb of city. The distance from the site of DHR to the boundary of city is 2 km, the radius of excluding area is 250 m.

REGULATION AND LICENSING

288
(CNEN-DR-RA-01/91)
Brazil-Argentina bilateral cooperation - Protocol 11. Nuclear safety and radiation protection. Visit of CNEA engineers to Brazil. Gasparian, A.E. (Comissao Nacional de Energia Nuclear (CNEN), Rio de Janeiro, RJ (Brazil). Dept. de Reatores). 2 Apr 1991. 39p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603561.

The activities carried out by Techniques of Argentine Organization (CNEA) during visitation to Angra dos Reis (Brazil) are related. Licensing procedures for nuclear installations (reactors), and transport, licensing of personnels for nuclear installations, quality assurance and regulatory inspections were discussed. (M.C.K.).

289
(IAEA-TECDOC-599, pp. 113-117)
Regulatory aspects of the use of PSA to evaluate technical specifications. Rumpf, J. (Staatiches Amt fuer Atomsicherheit und Strahlenschutz, Berlin (Germany, F.R.)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

Based on experiences gained in PSA activities the regulatory body of the GDR initiated a programme to investigate the feasibility of using PSA for the evaluation of technical specifications. This programme is just under work. In addition, to improve PSA, the GDR takes part in a programme which is aimed at performing plant specific level 1, PSA as well as and which enables operating organizations to carry out PSA on their own. The most important of some preliminary general findings presented in this paper are: - Technical specifications form a well established envelope of operational conditions and procedures. A total re-evaluation is not considered necessary; Probabilistic evaluation of technical specifications should be an integrated part of PSA activities (at least level 1). Single assessment is not considered reasonable; Probabilistic evaluation of technical specifications has to be based on plant specific information and realistic accident sequence calculations; Up to now no quantitative probabilistic criteria for technical specifications have been established. (author).

290
(IAEA-TECDOC-599, pp. 119-129)
Feasibility assessment of a risk-based approach to technical specifications. Atefi, B.; Gallagher, D.W.; Wohl, M.; Lobel, R. (Science Applications International Corp., McLean, VA (USA)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

To assess the potential use of risk and reliability techniques for improving the effectiveness of the Technical Specifications, the United States Nuclear Regulatory Commission (USNRC) initiated an effort to identify and evaluate alternative approaches that could bring greater risk perspective to these requirements. Among alternative approaches studied, a risk-based approach was chosen as the most promising for controlling plant operational risk using Technical Specifications. Technical and institutional issues associated with this approach were analyzed to assess the feasibility of implementing such an approach for determining Technical Specification requirements. Preliminary analysis shows that at this time there are no major obstacles to development of this approach. In order to further study all the practical issues associated with implementation of this approach, a pilot program would be useful. (author). 11 refs, 1 fig., 2 tabs.

291
(NUREG-0540-Vol.13-No.8)
Title list of documents made publicly available, August 1-31, 1991. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Freedom of Information and Publications Services). Oct 1991. 312p. OSTI; NTIS; INIS; GPO.

This document is a monthly publication containing descriptions of information received and generated by the US Nuclear Regulatory Commission (NRC). This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials, and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency. The following indexes are included: Personal Author, Corporate Source, Report Number, and Cross Reference to Principal Documents.

292
(NUREG-0750-Vol.33-Index2)
Indexes to Nuclear Regulatory Commission issuances, January-June 1991. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Freedom of Information and Publications Services). Oct 1991. 87p. OSTI; NTIS; INIS; GPO.

Digests and indexes for issuances of the Commission (CLI), the Atomic Safety and Licensing Appeal Panel (ALAB), the Atomic Safety and Licensing Board Panel (LBP), the Administrative Law Judge (ALJ), the Directors' Decisions (DD), and the Denials of Petitions for Rulemaking are presented in this document. These digests and indexes are intended to serve as a guide to the issuances. These information elements are displayed in one or more of five separate formats arranged as follows: Case name index, digest and headers, legal citations index, subject index, and facility index.

293
(NUREG-1440)
Regulatory analysis of proposed amendments to regulations concerning the environmental review for renewal of nuclear power plant operating licenses. (Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution). Aug 1991. 26p. OSTI; NTIS; INIS; GPO.

This regulatory analysis provides the supporting information for a proposed rule that will amend the Nuclear Regulatory Commission's requirements for environmental review of applications for renewal of nuclear power plant operating licenses. After considering various options, the staff identified and analyzed two major alternatives. Alternative

A is to not amend the regulations and to perform environmental reviews under the existing regulations. Alternative B is to assess, on a generic basis, the environmental impacts of renewing the operating license of individual nuclear power plants, and define the issues that will need to be further analyzed on a case-by-case basis. The findings of this assessment are to be codified in 10 CFR Part 51. The staff has selected Alternative B as the preferred alternative. 4 refs., 5 tabs.

294
(WHC-SA-1195)
Nuclear facility licensing, documentaion, and reviews, and the SP-100 test site experience. Cornwell, B.C.; Deobald, T.L.; Bitten, E.J. (Westinghouse Hanford Co., Richland, WA (United States)). Jun 1991. Contract AC06-87RL10930. 8p. (CONF-920104-2). OSTI; NTIS; GPO Dep. Order Number DE91014677.

From 9. symposium on space nuclear power systems; Albuquerque, NM (United States) (13-16 Jan 1992).

The required approvals and permits to test a nuclear facility are extensive. Numerous regulatory requirements result in the preparation of documentation to support the approval process. The principal regulations for the SP-100 Ground Engineering System (GES) include the National Environmental Policy Act, Clean Air Act, and Atomic Energy Act. The documentation prepared for the SP-100 Nuclear Assembly Test (NAT) included an Environmental Assessment, state permit applications, and Safety Analysis Reports. This paper discusses the regulation documentation requirements and the SP-100 NAT Test Site experience. 12 refs., 2 figs., 2 tabs.

295
(WHC-SA-1266)
Resolving the problem of compliance with the ever increasing and changing regulations. Leigh, H. (Westinghouse Hanford Co., Richland, WA (United States)). Jun 1991. Contract AC06-87RL10930. 9p. (CONF-920104-3). OSTI; NTIS; GPO Dep. Order Number DE91014680.

From 9. symposium on space nuclear power systems; Albuquerque, NM (United States) (13-16 Jan 1992).

The most common problem identified at several US Department of Energy (DOE) sites is regulatory compliance. Simply, the project viability depends on identifying regulatory requirements at the beginning of a specific project to avoid possible delays and cost overruns. The Radioisotope Power Systems Facility (RFSP) is using the Regulatory Compliance System (RCS) to deal with the problem that well over 1000 regulatory documents had to be reviewed for possible compliance requirements applicable to the facility. This overwhelming number of possible documents is not atypical of all DOE facilities thus far reviewed using the RCS system. The RCS was developed to provide a control and tracking of all the regulatory and institutional requirements on a given project. WASTREN, Inc., developed the RCS through various DOE contracts and continues to enhance and update the system for existing and new contracts. The RCS provides the information to allow the technical expert to assimilate and manage accurate resource information, compile the checklists, and document that the project or facility fulfills all of the appropriate regulatory requirements. The RCS provides on-line information, including status throughput the project life, thereby allowing more intelligent and proactive decision making. Also, consistency and traceability are provided for regulatory compliance documentation. 1 ref., 1 fig.

296
Rapport sur le controle de la surete et de la securite des installations nucleaires. Surete des installations. Securite et information (Report on nuclear installations safety and security control). Birraux, C.; Serusclat, F. Paris (FR); Assemblee Nationale (1990). 478p. (In French).

This report of the parliamentary office for evaluation of scientific and technological choices bearing on the safety and security of nuclear installations is divided into 2 volumes bearing on: - Volume I: nuclear installations safety. - nuclear safety and international organizations. - works separation: Finland, Belgium and Federal Republic of Germany. - French organization. - Volume II: security and information. - French nuclear security. - Public information.

297
Regulatory aspects of nuclear reactor decommissioning. Ross, W.M. pp. 164-174 of Decommissioning and demolition 1990. Whyte, I.L. (ed.). London (UK); Thomas Telford Ltd. (1990). 254p. (CONF-9004114-).

From 2. international conference on decommissioning offshore, onshore demolition and nuclear works; Manchester (United Kingdom) (24-26 Apr 1990).

Organised by Institution of Civil Engineers, North Western Association; University of Manchester Institute of Science and Technology; University of Manchester.

The paper discusses the regulatory aspects of decommissioning commercial nuclear power stations in the UK. The way in which the relevant legislation has been used for the first time in dealing with the early stages of decommissioning commercial nuclear reactor is described. International requirements and how they infit with the UK system are also covered. The discussion focusses on the changes which have been required, under the Nuclear Site Licence, to ensure that the licensee carries out of work of reactor decommissioning in a safe and controlled manner. (Author).

298
A survey of the Oyster Creek reload licensing model. Alammar, M.A. *Nuclear Technology (United States);* 93: No. 1, 47-52 (Jan 1991).

The Oyster Creek RETRAN licensing model was submitted for approval by the U.S. Nuclear Regulatory Commission in September 1987. This paper discusses the technical issues and concerns that were raised during the review process and how they were resolved. The technical issues are grouped into three major categories: the adequacy of the model benchmark against plant data; uncertainty analysis and model convergence with respect to various critical parameters (code correlations, nodalization, time step, etc.); and model application and usage.

299
Order concerning a nuclear reactor shutdown. *Umwelt- und Planungsrecht (Germany);* 11: No. 8, 320 (1 Aug 1991). (In German).

Judgment of the State Administrative Court of Baden Wuerttemberg in head notes including: The authority of the Minister-President to give general guidelines includes the right to issue single directives; in matters of prime political significance he can take measures to realize such aims. - It is no extraneous consideration for the supervisory board under atomic energy law to point out in an order concerning a nuclear reactor shutdown that the disallowed operation of a nuclear plant conflicts with the obligation of the state to provide protection and constitutes a penal offence. Further a discourse on the assignment of discretionary powers under Paragraph 19 Section 3 Clause 2 No. 3 of the Atomic Energy Law. (HSCH).

300
Experimental reactor BER-II. OVG Berlin, judgement of June 6, 1990, OVG 2 A 1.85. *ET, Energiewirtschaftliche Tagesfragen (Germany);* 41: No. 7, 478-480 (Jul 1991). (In German).

The OVG in Berlin dismissed an appeal against the first partial permit for an alteration to the experimental reactor BER II in Berlin. The plaintiff challenged - unsucessfully - the validity and basis of the so-called 30-millirem concept of the radiation protection law. The court also rejected a charge by the plaintiff that the permit was illegal because the reactor was not protected from air-craft crashes by a suitably equipped form of safety containment. (orig./HP).

301
Federal Administrative Court (BVerwG): Final decision on operating licence for the Brokdorf nuclear power station still pending. *Neue Zeitschrift fuer Verwaltungsrecht (Germany);* No. 8, 762 (1991). (In German).

Published in summary form only. BROKDORF REACTOR/operating licenses; OPERATING LICENSES/courts; LEGAL ASPECTS; COURTS

302
Official notice concerning the place and date of a public hearing to be held as part of the licensing procedure for the use of mixed oxide fuel elements in the Brunsbuettel reactor station. As of September 11, 1991. (Ministerium fuer Soziales, Gesundheit und Energie des Landes Schleswig-Holstein, Kiel (Germany)). *Bundesanzeiger (Germany);* 43: No. 185, 6972-6973 (2 Oct 1991). (In German).

Published in summary form only. BRUNSBUETTEL REACTOR/mixed oxide fuels; MIXED OXIDE FUELS/hearings; HEARINGS; REACTOR FUELING; LEGAL ASPECTS; LICENSING

ECONOMICS

303
(IAEA-TECDOC--607, pp. 15-30)
IAEA activities in the area of energy, electricity and nuclear power planning. Molina, P. (International Atomic Energy Agency, Vienna (Austria). Div. of Nuclear Power). May 1991. 275p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92601524. (CONF-8912153--).

From Workshop on electricity demand and supply planning in Europe, Middle East and North African countries; Nicosia (Cyprus) (11-15 Dec 1989).

In Experience in energy and electricity supply and demand planning with emphasis on MAED and WASP among member states in Europe, the Middle East and North Africa.

This paper describes the IAEA's activities in the area of energy, electricity and nuclear power planning and in providing assistance in this area to its Member States. Since this is a very broad area covering many different subjects, the description which follows mainly concentrates in those activities relevant to the purposes of the present workshop. Emphasis is given to the targets already achieved in terms of development of computer and analytical methodologies, providing training in the use of these methodologies and their application in the conduct of national planning studies. The prospects for future activities in this field are also reviewed. (author). 15 refs, 3 figs, 5 tabs.

304
(IAEA-TECDOC--607, pp. 193-206)
Problems related to energy and electricity demand forecasting and power generation in the Libyan Arab Jamahiriya. Aboughalya, E.D. (Ministry of Energy, Tripoli (Libya)). May 1991. 275p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92601524. (CONF-8912153--).

From Workshop on electricity demand and supply planning in Europe, Middle East and North African countries; Nicosia (Cyprus) (11-15 Dec 1989).

In Experience in energy and electricity supply and demand planning with emphasis on MAED and WASP among member states in Europe, the Middle East and North Africa.

This paper corresponds to some extracts of the results of a case study conducted jointly between the Libyan Energy Authority and an International Consultant with the objective to assess the economic viability of the nuclear power option within the wider framework of the expansion of the overall energy supply system for the country. The learnings of the process are reviewed in this paper and some of the principal results are illustrated at the end in graphical form. Major recommendations of the study are also summarized. A review of the situation of the supply and demand energy system of Libya is also discussed. (author). 7 figs.

305
(IAEA-TECDOC--607, pp. 259-272)
Experience in the use of WASP at the Societe Tunisienne de l'Electricite et du Gaz. Chakroun, C. (Societe Tunisienne de l'Electricite et du Gaz, Tunis (Tunisia). Direction des Etudes et de la Planification). May 1991. 275p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92601524. (CONF-8912153--).

From Workshop on electricity demand and supply planning in Europe, Middle East and North African countries; Nicosia (Cyprus) (11-15 Dec 1989).

In Experience in energy and electricity supply and demand planning with emphasis on MAED and WASP among member states in Europe, the Middle East and North Africa.

This paper presents what has been done at Steg using WASP. The implementation phase of the WASP and MAED models and the ENPEP Package is first described. Hence, the paper gives a list of the studies conducted by the Planning Department during the 80's, using WASP. The last WASP study, described in more details, deals with inter-fuel substitution in steg's power generation, for the next thirty years. Finally, some suggestions are proposed for WASP enhancement. (author). 1 fig.

306
(IAEA-TECDOC--610)
Financing of nuclear power projects in developing countries. (International Atomic Energy Agency, Vienna (Austria)). Jun 1991. 404p. (CONF-900932--). OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512.

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

This document is a summary of the "Topical Seminar on Financing of Nuclear Power Projects in Developing Countries, held in Jakarta between 4-7 September, 1990. The seminar presentations were divided into the following sessions: Keynote session (3 papers), Perspective of Nuclear and Fossil-fired Generation Costs (9 papers), Assessment of Problems and Constraints for the Financing of Large Power Projects, with particular Attention to Nuclear Power Projects (9 papers), Mechanisms for Financing Nuclear Power Projects in Developing Countries (11 papers). A separate abstract was prepared for each of these papers. Refs, figs, tabs and charts.

307
(IAEA-TECDOC-610, pp. 33-41)
Issues affecting the economies of nuclear power projects in the developing countries. Amrollahi, R. (Atomic Energy Organization of Iran, Teheran (Iran)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

Nuclear power, in its forty-odd years of history has gone through many peaks and troughs. However, it is regrettable to say that as yet it cannot be called a fully established industry since its evolution is still continuing. It is worth noting that a time span of almost half a century should have been sufficient to consolidate any industry no matter what the complexities. The reasons for this unfortunate state of affairs are wide and varied. Basically, such reasons fall into political, technical and economical categories and are discussed in this paper, with an emphasis on the economical and financial issues.

308
(IAEA-TECDOC-610, pp. 43-53)
Prospects of nuclear electricity generation in the ESCAP developing countries. Sudarsono, B.S. (United Nations Economic and Social Commission for Asia and the Pacific, Bangkok (Thailand). Natural Resources Div.). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The economies of developing countries depend to a greater or lesser extent on the economies of industrialized countries. Those in the ESCAP (United Nations Economic and Social Commission for Asia and the Pacific) region have shown remarkable progress in the past few decades, achieving sustained high growth rates in comparison with those of other regions. The paper discusses the performance of ESCAP economies, the energy consumption in support of economic growth, economic growth prospects and energy prospects. 5 refs, 9 tabs.

309
(IAEA-TECDOC-610, pp. 81-90)
Economics of nuclear power in Canada. Yu, A.M. (Atomic Energy of Canada Ltd., Mississauga, ON (Canada). CANDU Operations). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

By the end of 1990, the total installed generating capacity in Canada will be in excess of 104,000 MWe, of which 13% are CANDU nuclear, 56% hydro, 28% coal-and oil-fired, and the remaining 3% are diesel generators and combustion turbine capacities. Although Canada is blessed with abundant hydro potentials estimated at 188,191 MWe, only 46,084 MWe are thought to be economical and environmentally acceptable, most of which require long distance transmission lines to bring the power to the load centres. The commercialization of the CANDU system in Canada has demonstrated that CANDU is and will remain an economic and environmentally benign option to meet the current and future electrical energy needs of Canadians. This paper traces the evolution of the CANDU system in Canada, provides an analysis of the actual costs of nuclear as compared to other alternatives, and evaluates the long-term prospects of CANDU as an economical source of energy to supply the Canadian needs. Measures to extend the economic lives of CANDU reactors through rehabilitation and retubing programs will be discussed and their impacts on the levelized unit energy cost presented. The Canadian approach to irradiate fuel management and decommissioning will also be briefly discussed, and their impact on energy cost analyzed. In those provinces with abundant hydro electric potentials, an attempt is made to identify the timing whereby CANDU would become the economic choice. (author). 4 figs, 6 tabs.

310
(IAEA-TECDOC-610, pp. 69-80)
Economics of the USSR nuclear energy. Chernilin, Y.; Gagarinskij, A.; Tsurikhov, D. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (USSR). Inst. Atomnoj Ehnergii). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The current status of the USSR energy sector is analysed, with the role and the place of nuclear power in meeting energy demands being considered in detail. The results of analysing technical and economic aspects are described for both operating WWER nuclear power plants and WWER nuclear power plants under construction devoted to operate in once-through cycle, with additional safety and environmental aspects being paid serious consideration. Calculation of the cost of electricity generated by nuclear power plants is performed taking natural (land and water) resources used and compensation for possible accidents consequences into account. The main results of analysis are presented in percentage: Capital cost is about 43%; operational and maintenance cost is about 16%; fuel cost is about 23%; compensational cost of natural resources is about 17%; accident insurance cost is about 1%. Forecast is mad for the USSR nuclear power development up to year 2000. The necessity is shown to develop nuclear power in main industrial regions of the USSR from both economic and environmental viewpoint. (author). 5 tabs.

311
(IAEA-TECDOC-610, pp. 91-103)
Economic effects of nuclear power plant standardization in the Republic of Korea. Moon, Kee Hwan; Kim,

Seung Su. (Korea Atomic Energy Research Inst., Daeduk (Republic of Korea)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

In order to meet increasing electricity demand, due to rapid economic growth, additional constructions of nuclear power plant are expected in Korea where a large portion, about 50%, of the total electricity is currently being generated by the nuclear power plants. It is believed that standardization of nuclear power plants can make the construction and the management of nuclear plants more economical and efficient. In Korea, the standardization is also expected to help indigenousness of technology and to improve safety of nuclear power plants and load factor. In this study, Yonggwang 3,4 units are selected as a reference plant and the economic effects of standardization on capital cost are analyzed for this reference plant. The economic effects are estimated separately in two parts, i.e., overnight cost and time-related cost. For the purpose of estimation on economic effects, this paper employs two case studies reflecting the experiences of Japan and France. The reason for carrying out the two case studies is to consider the uncertainties of the economic effects which can be caused by little experience of standardization in Korea. The result of this study shows that the total capital cost can be reduced at least by 10.6% due to the savings of time-related cost. If the reduction of overnight cost is considered together, then the total capital cost can be lowered at least by 21.7%. Thus, it is clear that standardization of nuclear power plants demonstrated great contribution to the savings of capital costs. (author). 15 refs, 2 figs, 9 tabs.

312
(IAEA-TECDOC–610, pp. 105-123)
Capital cost reduction of nuclear power plants. Noda, H. (Tokyo Electric Power Company, Inc., Tokyo (Japan). Nuclear Power Design Div.). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

Extensive efforts to develop nuclear power have been continuously made by Tokyo Electric Power Company (TEPCO) based on a policy of power supply diversification. TEPCO has 12 BWRs in operation, 3 BWRs under construction, 2 ABWRs (Advanced Boiling Water Reactors) in preparation for construction. This report describes the history and fruits of BWR and ABWR development and the capital cost reduction of them. The capital cost reduction is basically achieved by the adoption of design standardization, rationalization of design, replication merit and shortening of construction period. The ABWR, an optimum design for the 1990s and beyond, is also introduced in terms of both performance and economy. (author). 13 figs, 4 tabs.

313
(IAEA-TECDOC–610, pp. 125-136)
New nuclear economics: Affordable electric power. Bruschi, H.; Henderson, R. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Nuclear and Advanced Technology Div.). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

A new, simplified pressurized water reactor design and plan for construction results in a drastic reduction in plant capital cost and a significant improvement in overall power costs. The design simplification has enhanced the overall safety of the plant while eliminating 60 percent of nuclear island valves, 35 percent of total plant pumps, and 75 percent of nuclear island piping. These and other reductions yield a 30 percent reduction in overall capital cost compared to a conventional nuclear plant, for a power output tailored to the electric grid size of developing countries. The construction schedule is also shortened significantly, with the use of modular techniques developed in the shipbuilding industry. A construction schedule of 36 months is achievable with this new approach. Further, the simplified plant design enhances maintainability and thus reduces costs of operation and maintenance (O and M). The large reduction in capital cost, combined with the shorter construction schedule, reduced O and M cost, and lower fuel costs, produces a power cost for the new design which is very competitive with alternative forms of generation. This paper illustrates the new nuclear economics by comparing the costs of the Westinghouse AP600 with those of alternative forms of power generation and the standard Westinghouse two-loop 600MW plant. The AP600 plant is being developed under sponsorship of the U.S. Department of Energy and the Electric Power Research Institute. (author). 8 figs, 8 tabs.

314
(IAEA-TECDOC–610, pp. 137-145)
Economic importance of financing export oriented nuclear power projects. Lehmann, G.; Fourre, J.P.; Plante, J.; Grentz, H.J. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Paris-La-Defense (France)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

How to define a financial structure suitable to the project: The breakdown between the available resources allocated to the project and the portion being financed; the requirements for financing the imported portion and the locally made portion of a contract must be treated separately; the selection of the financing currency with regard to the contract currency; the setting up of financing reserves for contingency cost overrun; the effect of the contract structure of the project; the repercussion of interest rates on the competitive situation of nuclear energy; macro-economical consequences of hard currencies payments financing. (author). 3 charts.

315
(IAEA-TECDOC–610, pp. 147-154)
Evaluating investment costs and economic performance criteria for nuclear power in Romania. Sima, C. (Institute of Power Studies and Designs (ISPE), Bucharest (Romania)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The transition from an overcentralized economy to a rational and efficient one asks for the implementation of a judicious economic policy which should cope with the specific conditions of Romania. The present work aims at a thorough analysis of several major aspects related to the evaluation and setting up of the economic efficiency in the power field, and to the implication possibilities into the market economy. (author). 1 fig.

316
(IAEA-TECDOC-610, pp. 155-167)
Prospects of nuclear power in Java. Sudja, N. (State Electricity Corp., Jakarta (Indonesia). Div. of System Planning). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

In view of the present low level of per capita energy consumption, Indonesia's potential for load growth in the future is very high. This coupled with the large size of the population, the currently known indigenous resources such as coal, hydroelectric, oil, natural gas and geothermal which presently appear abundant will soon be depleted. Therefore, Indonesia has to look at the nuclear option. In the initial stage of rapid electricity development in Indonesia, PLN (State Electricity Corporation) depended largely on oil-based generating capacity. However, in the late 1970s, PLN committed to reduce its reliance on oil towards more economical use of fuel such as hydroelectric, geothermal, coal and natural gas. Although, utilization of a nuclear power plant has been discussed since the late 1960s, its role in PLN's Investment Program has never been defined. This paper is based on PLN study of generating cost of future base-load unit alternatives for the Java Power System. The objective of the study is to improve the understanding of the competition among those units, by providing an analysis and sensitivity studies of major assumptions such as investment costs, fuel costs and discount rate. And at the last part, this paper presents the general problems of financing and review and analysis of liability involved in nuclear power project. (author). 8 refs, 11 tabs.

317
(IAEA-TECDOC-610, pp. 171-185)
Project financing in the energy sector: The Indonesian experience. Hutapea, R.O. (Ministry of Mines and Energy, Jakarta (Indonesia)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

Past Indonesian experiences in arranging Project Financing or Limited Recourse Financing in the Energy Sector and the feasibility of such a structure for Nuclear Power Projects are reviewed in this paper. Many of the projects in the energy sector, including Nuclear Power Project are capital intensive projects. A single project may require an investment level of several billion dollars. Large capital investments of this nature may drastically limit the availability of funds from external sources used to finance other non-energy projects in a developing country such as Indonesia. Therefore, the financing concept of a project is becoming a paramount element in the policies adopted by the Republic of Indonesia. As an example, a non-recourse finance structure is the preference of the government for export oriented energy projects. The aim of the government financing policy for export oriented energy projects is to limit the recourse to the applicable Indonesian state enterprise (the project sponsor) by transferring to third parties as much of the project risks as possible. Most of the export oriented energy projects come close to achieving a non-recourse structure. Domestic oriented energy projects however, continue to be financed with a loan guarantee by the government or by state owned enterprises. In addition to maintaining fund availability from external sources for the non-energy sector, Indonesia's reported external debts as well as debt service ratios are other factors in considering a limited recourse financing structure for capital intensive projects. (author).

318
(IAEA-TECDOC-610, pp. 213-217)
Financing a nuclear power plant in a developing country: Experience with the DAYA BAY nuclear plant, China. Lavril, F.X. (Banque National de Paris, Paris (France)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The project: A 2 x 900 MW nuclear power plant located a few miles from Hong Kong which shall import 70% of its production; Most of the contracts of the Nuclear part of the DAYA BAY project have been signed with French companies; The important part of Electricite de France (EDF) in the project due to its experience as a nuclear electricity producer. The main contracts and their financing: EDF wins the contract for technical assistance, for completion realization and putting into service of the plant; Framatome wins the contract for the supply of the two nuclear islands; The financing of the contracts in accordance with the OECD nuclear consensus. The additional contracts and their financing: The civil works contract (Campenon Bernard); The erection contract (Framatome Spie Batignolles); The financing of such contracts at usual conditions instead of the OECD nuclear consensus conditions. Conclusion: How to get the best financing for a nuclear project. (author).

319
(IAEA-TECDOC-610, pp. 219-236)
Financing of nuclear power programmes in developing countries: The Indian experience. Basu, R. (Nuclear Power Corp. of India Ltd., Bombay (India)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The paper states the objectives of the Nuclear Power Programme in India and the strategy being pursued therein. Keeping in mind the history of nuclear power in India, the nature of technology involved, the risks and the uncertainties associated with the development of nuclear technology in developing countries, the paper specifies an appropriate organizational framework and a financing structure within which the full potential of nuclear power is expected to be realised. The paper looks at projections of capital investment required for the period 1989-2001 A.D. i.e. till completion of the first stage of the nuclear power programme in India. The projections examine nuclear power costs in India over the last three decades or so. It is seen that the capital costs as well as operation and maintenance costs have increased with time. The capital investments required for the programme are estimated at about Rs.15,755 crores at 1989-90 prices, based on typical "specific" capital costs of Rs.19,000 - 20,000 per KWe installed. The paper discusses the principal sources of funds for the same which include equity contributions, internal surpluses and market borrowings. It also elaborates a financial plan which takes into consideration the optimal and likely mix of these. It is seen that in a realistic scenario, to meet a shortfall of Rs.3,305 crores, borrowings have to be of the order of Rs.23,655 crores out of a total fund requirement of Rs.37,770 crores. The paper finally examines some alternative financing schemes such as joint-ventures amongst suppliers/vendors and the utility, in an attempt to share the financial burden of the large investments planned. (author). 1 fig., 10 tabs.

320
(IAEA-TECDOC-610, pp. 237-242)
Financing the export of nuclear power plants to developing countries. Fujii, K. (The Export-Import Bank of Japan, Tokyo (Japan)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

Japanese manufacturers of nuclear power plants are confident in quality and reliability of their products but so far they have played a minor role in the international market. In studying the finance for the future nuclear power projects in the developing country, the financial institutions will take into consideration such characteristics of the nuclear power project as: (1) Public acceptance and safety; (2) Commitments by host government and its involvement which are internationally required; (3) Huge amount of capital requirements over a long term for commissioning, operation and maintenance; (4) inability of the domestic market oriented project to generate foreign currency earnings; (5) Long lasting project life; (6) Creditworthiness of both the project owner and the host government and the debt problem of the country; (7) Priority; (8) Feasibility; (9) Available engineers and their level of technology, etc. Under the present circumstances, because of the inherent nature of the project and the creditworthiness, the owner of the project who imports the nuclear power plant and the host government will find it very difficult to receive enough funds for the project from the financial institutions, including the export credit agencies, and the manufacturers (exporters). On condition that the owner of the project and the host government make every effort and necessary arrangement to materialize the project, one of the measurers for the creditors to share such burdens and risks will be to form an international consortia among manufacturers and to organize co-finance among ECAs, private financial institutions and Multilateral Development Banks. (author).

321
(IAEA-TECDOC-610, pp. 243-256)
Egypt's nuclear power programme: Economic and financial constraints. Abdel Gawad, A.S.; Mekhemar, S.S.; Zaghloul, K.S. (Atomic Energy Authority, Cairo (Egypt)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

Egypts' needs for electric power up to the year 2002 is assessed. The potential of different energy resources in Egypt is evaluated. The nuclear option is discussed from economic, social, and technological point of view. The special situation of Egypt as a developing country with weak infrastructure and limited qualified manpower is stressed. Advanced nuclear technology is discussed from the perspective of its appropriateness to the needs and conditions prevailing in developing countries. Some national and international resources to finance nuclear power projects were proposed and the potential of each was assessed. (author). 11 refs, 2 figs, 6 tabs.

322
(IAEA-TECDOC-610, pp. 257-264)
Nuclear power plant programme financing: The EDF experience. Fiancette, G.A. (Electricite de France (EDF), 75 - Paris (France)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The French nuclear programme, which has evolved over the past 15 or so years, has had very substantial financing needs. These have led E.D.F. to become heavily indebted. The burden of this debt has led the company to organize its financial strategy in such a way as to keep the situation tolerable. Now E.D.F. is beginning to pay off its debt. Companies which appear to find themselves in a different environment should find in an analysis of the conditions leading to this success a source of instruction to reflect upon. (author). 1 ref., 7 figs, 1 tab.

323
(IAEA-TECDOC-610, pp. 265-271)
Experience in the financing of nuclear power projects. Herold, H. (Kreditanstalt fuer Wiederaufbau, Frankfurt am Main, (Germany, F.R.)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932-).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

KfW has gained manifold experience in the granting of export loans to finance supplies and services for nuclear power plants in developing countries. The main problem KfW had to face were the considerable delays in the project execution due to administrative and financial difficulties which, apart from the negative impact on the national economy, resulted in higher project cost. So KfW had to adjust its financing accordingly. But the financing of foreign exchange costs did not pose very serious problems; what proved to be more difficult was the financing of local costs due to shortage of government funds and constraints in local capital markets. Thus it is vital to secure also the provision of sufficient local finance. The development of functioning domestic capital markets is particularly important for the funding from sources within the buyer country. The foreign currency financing of local costs tends to increase the debt burden and carries a foreign exchange risk. From the angle of fostering a country's technical and economic development, it would be desirable to make local finance to that extent available that the largest possible local content of the project can be realised. Both the execution and the financing of nuclear power projects are difficult by nature because of the long implementation period involved and the huge financing amounts. As a consequence the financing structure of such sophisticated projects should be rather simple and conventional in order to offer enough flexibility for adaptation in the event of unforeseeable changes during the project implementation period. (author).

324
(IAEA-TECDOC–610, pp. 273-275)
French assessment of the OECD consensus. Thuillier, J.P. (Ministere de l'Economie, des Finances et du Budget, Paris (France). Direction des Relations Economiques Exterieures). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

On July 31, 1984, OECD countries have agreed on special guidelines for the financing of nuclear power plants exportations. Such rules may seem drastic for developing countries keen to diversify their energetic resources. This paper discusses the reasons why the chances of a changing of the rules are slim, and thereafter, the more flexible attitudes an exporting country like France would be able to adopt in a way to better-up the conditions of installation of nuclear power stations in developing countries. (author).

325
(IAEA-TECDOC–610, pp. 279-284)
Estimated capital requirements for electricity generation in developing countries and a discussion of financing options. Eschenberg, H.H.H. (Bourne and Co., Charlotte, NC (USA)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

If developing economies are to realize or even come near their growth targets, they must continue to invest in the expansion of energy capacity. Serious estimates of capital requirements for the expansion of electric power supply in them are of the order of US$60 - US$100 billion per year of which about US$25 billion would be in foreign exchange. Prospects for increased capital flows in the 1990's are slim because of emerging capital need for reconstruction of the economies in Eastern Europe and the Soviet Union, and sharply reduced commercial banks' lending to developing countries. Also domestic public resources have been and continue to be limited because of competing claims for social and economic needs. The effects of a shrinking capital pool have exacerbated the risk perception of potential investors of large, capital intensive, energy projects with long gestation periods. Nuclear projects have been doubly vulnerable, because of their uncertain gestation periods and the additional perceived safety risk. Recognizing the growing financing constraints, this paper discusses that capital mobilization efforts, especially for nuclear power, will have to extend beyond traditional boundaries, that resources mobilization for nuclear power from hitherto not available multilateral sources would be most helpful and finally, that domestic capital markets will need to be tapped. (author).

326
(IAEA-TECDOC–610, pp. 285-298)
Chinese nuclear power projects investment programming model and analysis of financing approaches. Li Tian. (China Inst. of Nuclear Industry Economics, Beijing (China)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

China nuclear power projects investment programming model (NUPIP) is presented in this paper. Different kinds of financing approaches are analysed. The annual financing requirement is calculated according to a supposed nuclear power programme from 1991 to 2000 which may be the maximum financing amount scenario in this period. Sensitivity analyses are made for some of the variables which are of effect on financing requirement. The results show that the government investment and funds raised by localities are the principal financing resources for home-made units at the initial stage of nuclear power development, and the reserve funds from sales profits of nuclear electricity will become the important resource along with the development of nuclear power. (author). 3 tabs.

327
(IAEA-TECDOC–610, pp. 299-313)
Financial feasibility of small and medium nuclear power plants: A case study for Egypt. El Saiedi, A.F.; Woite, G. (Nuclear Power Plants Authority, Cairo (Egypt)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

Financing problems are a major constraint on more widespread use of nuclear power, in particular in developing countries. One problem is the sheer magnitude of the investment cost of large nuclear power plants. Small and medium power reactors would have lower absolute costs and would offer some additional advantages. Taking Egypt as an example, the paper addresses the issue to what extent the features of small and medium power reactors designs could improve the financial feasibility of nuclear power plants in developing countries. (author). 6 refs, 4 figs, 4 tabs.

328
(IAEA-TECDOC–610, pp. 315-324)
Countertrade and export of electricity in financial planning. Fabijancic, A.; Subasic, D. (Association of Utility Organizations of Croatia, Zagreb (Yugoslavia)). Jun 1991. 404p. OSTI; NTIS

(US Sales Only), INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries

Preparation for continuation of nuclear program in Yugoslavia, resulted in international invitation of bids for construction of a series of nuclear power plants in Yugoslavia, the bids collection for nuclear fuel cycle and nuclear power plant technology selection as well as the commercial bids collection for NPP Prevlaka equipment and services supply being the first plant from the series. Financing plan for NPP Prevlaka, the role and importance of countertrade combined with the export of electricity in financial planning are set out in the paper. In spite of nuclear program termination in the middle of 1989, by enacting a moratorium on nuclear power plants construction, up to that moment in the course of financial planning several approaches were introduced. From that experience, the basis for current strategy of financing the power plants in Republic of Croatia has been created. (author). 10 refs, 2 tabs.

329
(IAEA-TECDOC–610, pp. 325-330)
Nuclear development potential in new Brunswick: The "Whole-to-coal" model. Little, K.B. (New Brunswick Power, Fredericton, NB (Canada)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries; Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

New Brunswick Power is a small Canadian utility with a single 635 MW CANDU nuclear unit at Point Lepreau. The high capital and early year power costs of the nuclear option are difficult for a smaller utility to cope with, even in a developed country. Recent discussions with Atomic Energy of Canada Limited in respect of early construction of a demonstration CANDU-3 unit (450 MW) led to the development of a "whole-to-coal" concept. This model limits the initial capital investment and the early-year power costs for the buying utility to the level of an equivalent coal plant, thereby making it easier to finance and absorb a new nuclear unit. Elements of the model may be of interest for nuclear power programs in developing countries. (author).

330
(IAEA-TECDOC–610, pp. 331-335)
Nuclear power: Financing options for developing countries. Groom, G.W. (Westinghouse Electric Corp., Pittsburgh, PA (USA)). Jun 1991. 404p OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries, Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries

The Westinghouse AP600 nuclear power plant is a new simplified pressurized water reactor design that results in a dramatic reduction of plant capital cost and significant improvement in overall power generation costs. A major benefit that adds to the economy of the new AP600 design is the shortened construction schedule. This, along with a 30% reduction in capital costs, allow finance costs related to an AP600 project to be substantially contained and provides more flexibility in financing vehicles used during the construction program and subsequent repayment term. The shortening of the construction schedule coupled with changes in U.S. Eximbank programs will allow for the utilization of short term funding vehicles during the construction phase that will add to the already significant interest cost savings. This paper will detail a comparison between the financing costs of previous conventional nuclear power projects and the new AP600 design and address funding mechanisms that can be utilized to provide the most competitive financing available for the project. (author). 3 figs, 2 tabs.

331
(IAEA-TECDOC–610, pp. 361-368)
The BOT model for nuclear power plant projects from a supplier's point of view. Ruess, F.; Lebreton, G. (Nuclear Power International, Paris-La-Defense (France)). Jun 1991. 404p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603512. (CONF-900932–).

From International seminar on the costs and financing of nuclear power in developing countries, Jakarta (Indonesia) (4-7 Sep 1990).

In Financing of nuclear power projects in developing countries.

The significant financing requirements in particular for Nuclear Power Plants as well as the mandatory need for a professional management of their construction and operation has led to the discussion of so called "BOT" models for the implementation of such projects Under this model, a "Joint-Venture" would be established which would act as the Purchaser, owner and operator of the Plant. The equity of this Joint-Venture would be held by the Supplier(s) of the Power Plant, the local private or state-owned utilities and possibly additional private or institutional investors. The Joint-Venture would, in addition to the Shareholder agreement concluded between the investors, enter into a Power Sales Agreement with the local utilities, a Supply contract with the vendor of the Power Plant and Loan Agreement for the financing of the Project. After a defined period, during which the return on and of equity of the investors would be affected and the loans taken out for the project would be repaid, the ownership of the Plant would be transferred to the local utility. The paper briefly describes the structure of such a model and discusses in particular the assessment of its feasibility from a supplier's point of view. It is concluded that any private investor including the supplier of the plant will necessarily limit his financial obligations in such a scheme to his equity in the joint venture. Hence, the feasibility of such a model depends on the security concept for the financing of the project which according to our present experience will have to comprise an exposure of the Government of the country, in which such a project shall be implemented. (author). 1 fig.

332
Economic perspectives of nuclear power: Review and outlook. Hansen, U (Verein Deutscher Ingenieure (VDI) - Gesellschaft Energietechnik, Duesseldorf (Germany)). pp. 241-268 of Nuclear power: Today, tomorrow. Duesseldorf (Germany), VDI-Verl. (1991). 323p (In German). (CONF-9103202–).

From Conference on nuclear energy: Today, tomorrow; Aachen (Germany) (18-19 Mar 1991).

The decision for stepping up nuclear power plant capacity in the Federal Republic of Germany proved to be a success in economic terms, as revealed by the present value of expenditure for existing power plants. A possible alternative, namely enhanced use of inland coal for electricity production, would have increased the expenses by about DM 160 thousand millions, and a mixed strategy of enhanced use of coal and nuclear energy, as defined in the 'Jahrhundertvertrag' would have required additional expenditure of DM 90 thousand millions. Only the alternative of using the cheaper coal imports keeps the costs at comparable level with nuclear energy. On the energy market, a substitution of fossil energy sources by one another can be observed, creating an equivalent price basis, depending on caloric value and differences in treatment. In electricity generation, coal, oil and gas compete with nuclear energy. All conventional alternatives for electricity generation currently are equal in costs. (orig./HP).

333
The global mission of nuclear power. Neumann, J. *Jaderna Energie (Czechoslovakia);* 37: No. 5, 195-198 (May 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

The contribution of nuclear power to satisfying the future energy needs of mankind and to alleviating the greenhouse effect problem is discussed. It is concluded that in addition to fossil fuels and the hydro-energy, nuclear power is the only macroeconomic source of energy for the majority of countries in this and the next centuries. In the first decade of the 21th century the production capacity of nuclear engineering shall roughly double, and high-temperature and fast-breeding reactors shall play an important role. It is expected that the research into nuclear fusion will progress. (Z.M.). 5 figs., 4 tabs., 8 refs.

334
The role of nuclear power in the reassessment of Czechoslovakia's energy policy. Cibula, M. *Jaderna Energie (Czechoslovakia);* 37: No. 7, 276-278 (Jul 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at USD 10.- per typewritten page.

The role of nuclear industry in an effective solution of Czechoslovakia's economic, energy and ecological problems is discussed. It is concluded that the impacts of slowing-down of the construction of nuclear power plants can only be overcome by extending the operation of the ecologically unfavorable coal-fired power plants; orientation either to the construction of natural gas-fired power plants with combined steam-gas cycles associated with the use of heat, or to electricity imports does not offer a fundamental solution to the above problem. (Z.M.). 5 refs.

335
On Czechoslovak energy policy. Trnka, J. *Jaderna Energie (Czechoslovakia);* 37: No. 6, 225-227 (Jun 1991). (In Czech).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at US$ 10 per typewritten page.

The principles are discussed of the new Czechoslovak energy policy, whose aim it is to enable an energy-saving development of the society and thus to contribute to increasing the standard of living and to a healthier life of the population. This is conditional on structural changes in and innovations of technologies, products and facilities in the whole national economy, including the power industry. Among the most important structural changes in the latter, already implemented, was the switching to nuclear power as the main source of electricity. Since 1982, all electricity generation capacity increases in condensation power plants have been covered by nuclear power plants. By the end of 1990, 155 TWh of electricity had been generated by nuclear power plants; this would require mining and burning of about 170 million tons of brown coal if the same amount of electricity were to be produced by coal-fired power plants. (Z.M.).

336
Politics of electricity production. Price, T. *Nature (London) (United Kingdom);* 351: No. 6326, 435-436 (6 Jun 1991).

Almost everywhere the antinuclear lobby remains entrenched and vigorous. At the same time nuclear power makes about 17 per cent of the world's electricity - as much as came from all sources in 1957. Ten of the 25 countries with operating reactors make more than a third of their electricity this way. This disparity between perception and reality must be dealt with before nuclear power can make its proper contribution in a world whose population will be 60 per cent larger by 2025. Even allowing for conservation and efficiency, 50 per cent more energy will be needed by then. It cannot come on that scale from the renewables. Only coal, gas, oil, hydro and nuclear power produce really big packets of energy. The first three have the greenhouse question mark hanging over them, while hydro resources are limited. Reductions in the emissions of carbon dioxide, already promised by Europe and Japan, would be impossible without a full nuclear contribution. There is scope for at least a doubling of nuclear electricity over the next 30 years. But today the nuclear construction industry's order books are almost empty. The problems are almost entirely political: before the politicians will act, the public's concerns about economics, safety, waste disposal and nuclear proliferation must first be allayed. The background to the creation of a more favourable nuclear climate is discussed. (author).

MISCELLANEOUS

337
(IAEA-TECDOC--599, pp. 63-76)
Optimization of technical specifications by use of probabilistic methods - A Nordic perspective. Laakso, K.; Engqvist, A.; Knochenhauer, M.; Kosonen, M.; Liwang, B.; Mankamo, T.; Poern, K. (Valtion Teknillinen Tutkimuskeskus, Espoo (Finland)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379--).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

The Technical Specifications of a nuclear power plant specify the limits and conditions for plant operation from the safety point of view. These operational safety rules were originally defined on the basis of deterministic analyses and engineering judgement. As experience has accumulated, it has proved necessary to consider problems and make specific modifications in these rules. Developments in probabilistic safety assessment have provided a new tool to analyse, present and compare the risk effects of proposed rule modifications. The main areas covered in the project are operational decisions in failure situations, preventive maintenance during power operation and surveillance tests of standby safety systems. This project is part of the Nordic safety programme 1985-89 sponsored by NKA, the Nordic Liaison Committee for Atomic Energy. The work has been financed in part by the Nordic Council of Ministers and in part by the participating Swedish and Finnish institutions, power companies and regulatory bodies. (author). 3 refs, 6 figs, 1 tab.

338

(IAEA-TECDOC–599, pp. 77-87)
EPRI perspectives on the use of risk-based technical specifications in controlling plant operations. Sursock, J.P.; True, D. (Electric Power Research Inst., Palo Alto, CA (USA)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

In recent years there has been considerable interest in the use and application of probabilistic risk techniques to the development of technical specifications. The Electric Power Research Institute (EPRI) has participated in the investigation and application of these methods in the U.S. and continues to support new and innovative approaches. This paper describes the program EPRI has established for the evaluation and development of risk-based technical specifications in controlling plant operations. The paper identifies institutional and technical obstacles associated with the concept of a real-time risk monitor, and then proceeds to describe the current EPRI program. Flexible technical specifications actions or "flex specs" promise to increase operating flexibility by providing the plant operating staff with pre-planned alternative actions to be taken in response to a specific limiting condition for operation, depending upon the plant configuration. The introduction of additional complexity into the technical specifications will require additional tools to be developed to assist the plant operating staff in determining the options available. EPRI is investigating a PC-based tool for use in this application. This tool, called an Integrated Risk Advisor (or IRA), will provide the control room operators with information on flex spec options, support system unavailabilities, tracking of component and system reliabilities, as well as access to standard technical specification information. (author).

339

(IAEA-TECDOC–599, pp. 89-95)
Status of PSC and technical specifications improvements based on probabilistic methodology. Volkovitskij, S. (USSR State Committee for Supervision of Safety in Industry and Nuclear Power, Moscow (USSR). Science and Engineering Center for Safety in Industry and Nuclear Power). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

In 1990, three supporting probabilistic indicators were included in the new version of the USSR main regulatory document "General Rules of Ensuring Nuclear Power Plant Safety". The series of guidelines for conducting PSA is under development. The nuclear regulatory body encourages the practical use of PSA methodology both for NPP design and operation. Two examples of the use of probabilistic methodology for technical specifications assessment are described. It is stressed that the regulatory body considers probabilistic methods as an important but supporting tool for making regulatory decisions. (author). 2 refs, 1 figs.

340

(IAEA-TECDOC–599, pp. 97-112)
Allowable outage times (AOTs) and surveillance test intervals (STIs) reevaluation by PRA procedures. Serradell Garcia, V.; Martorell Alsina, S.; Verdu Martin, G.; Vazquez, M.T.; Calvo, J.I. (Universidad Politecnica de Valencia, Valencia (Spain). Dept. de Ingenieria Quimica y Nuclear). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

In the early 1980s, tools for allowable outage times (AOTs) and surveillance test intervals (STI) evaluations by PRA procedures were developed. Most have been implemented into program codes. Some were developed before 1980 to assess Plant Safety from a risk point of view (PRA level 1). Objective of the paper is to show how these tools can be used in an AOT and STI evaluation program. An analysis scheme is proposed, stressing the most important topics related to qualitative and quantitative analysis. In the last one, time-dependent or independent risk evaluation has been considered separately. Fault and event trees are obtained through qualitative analysis, and minimal cut set generated for use in quantitative analysis. By means of time-independent quantitative analysis a time-independent risk estimation is obtained. The most important STI and AOT requirements are identified. Also, some sensitivity and uncertainty analysis are performed. The time-dependent analysis uses the results from previous qualitative and quantitative studies. The analysis is specially useful to accomplish with AOT and STI re-evaluation because of the time-dependence of these requirements. Additional sensitivity analysis lead to review test and maintenance influence on risk, in order to confirm results from AOT and STI evaluation and are related to: hypothesis and models, data, human error and common cause failures. At the end of this paper a case of application with the corresponding results of whole analysis is presented. The case of application analyses the benefits of the alternate strategy testings: staggered or sequential for various surveillance test intervals. Furthermore, additional calculations were performed to investigate the sensitivity fo the results to the input used. In particular, we study the impact on system unavailability when the time-related (standby) failure fraction varies from 0 to 1 (all functional failures) are demand-related.

341

(IAEA-TECDOC–599, pp. 131-138)
Approaches for ascertainment of allowable outage times (AOTs). Theiss, K. (Technischer Ueberwachungs-Verein Norddeutschland e.V., Hamburg (Germany, F.R.)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order

Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

On the background of the requirements of German Nuclear Safety Criteria of NPP the KTA-report 1407 was established as a guideline to represent methods concerning the ascertainment of allowable outage times of safety systems during NPP operation. The methods described are based on both probabilistic and deterministic approaches and have been used in former times in licensing procedures of NPP. (author). 2 refs, 2 figs, 2 tabs.

342
(IAEA-TECDOC-599, pp. 139-151)
WWER plant probabilistic safety assessment. Volkov, V.A.; Larin, E.P. (All-Union Research Inst. for Nuclear Power Plant Operations (VNIIAES), Moscow (USSR)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

This paper addresses the ways of probabilistic safety assessment for WWER-1000 reactors in operation. Probabilistic analysis is supposed to be used to optimize operational documentation, to assess weaknesses and enhance safety and efficiency of the operating plants. Probabilistic safety assessment (PSA) is a necessary part of safety justification for NPPs both operating and under design. To practically implement the task much effort is needed to process information, prepare computers codes, to do calculations and adopt the results. Operational experience shows that WWER plants can be improved by improving stability, reducing the number of shutdowns, labour consumption as well as improving operational documentation and taking account of all hypothetical transients. The best possible way to achieve this goal is to use PSA. In this paper PSA performance programme adopted in VNIIAES is described. (author). 6 figs, 1 tab.

343
(IAEA-TECDOC-599, pp. 153-163)
Uncertainty analysis in the process of reliability estimation. Holy, J. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

This paper deals with the uncertainty analysis as an important line of reliability analysis. In the first part of material the framework of uncertainty studies performed in NRI Rez is provided (fault tree method, Lognormal distribution, error factor, modularization). In the second part of this one the interesting general results (behaviour of error factor) are published, studied and commented. An original attempt at time dependent uncertainty analysis is presented. (author). 3 refs, 1 fig.

344
(IAEA-TECDOC-599, pp. 165-175)
The use of probabilistic safety analysis methods for planning the maintenance and testing unavailabilities of essential plant at Heysham 2 AGR power station. Horne, B.E. (Nuclear Electric plc, Barnwood (UK). Technology Div.). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

The Paper describes the development of the practice adopted at Heysham Power Station for the control of the removal of essential plant for maintenance and testing. This has been based on the definition of two operational categories derived from probabilistic and deterministic fault criteria, and used an advanced interactive computing facility to demonstrate compliance with these criteria. This facility, the Essential Systems Status Monitor (ESSM), contains a "living model" of possible failure modes of the essential systems which is continuously updated by the operator as plant is removed for planned maintenance and testing and as systems are reconfigured. The ESSM is also used to plan plant outages so as to minimise the operational risk during maintenance. The overall strategy adopted at Heysham 2 has resulted in simple operating instructions and increased flexibility in planning plant outages for maintenance and testing. (author). 2 refs, 6 figs, 2 tabs.

345
(IAEA-TECDOC-599, pp. 177-179)
Methods of evaluation and service reliability of unique devices and plants: Summary. Fedik, I.I.; Golubev, M.P. (Scientific Industrial Association "Lutch", Podol'sk (USSR)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

In the present report the results of approbation and choice of methods of probability analysis of service life and reliability estimation of unique products, nuclear power plants are given.

346
(IAEA-TECDOC-599, pp. 253-269)
Operational decision alternatives in failure situations of standby safety systems. Mankamo, T.; Kosonen, M. (Avaplan Oy, Espoo (Finland)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379-).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

When a failure is detected in safety systems during plant operation, the risk level may increase much above the baseline, specially in rare multiple failure situations. In such cases the operators face different operational alternatives, eg. testing the remaining parts of safety systems, and/or decision on plant shut down or some backup arrangements. A series of applications at the Finnish nuclear power plants prove that the probabilistic risk and decision analyses can provide support for the systematic comparison of these alternatives. At the TVO plant (BWR), the probabilistic analysis has shown that in failure situations of the residual heat removal systems, the shutdown constitutes a higher risk than continued operation over usual repair times of less than one day. Based on the results, appropriate modifications to the technical specifications and operating instructions are under way concerning repairs of multiple train failures in residual heat removal systems during power operation. (author). 12 refs, 8 figs.

347
(IAEA-TECDOC–599, pp. 207-217)
NPP channel structure safety system reliability analysis. Methods and computer code SHARM-2. Polyakov, E.F.; Shiverskij, E.A.; Loskutov, G.Yu. (Reserach and Development Inst. of Power Engineering, Moscow (USSR)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

Special investigations on the methods for reliability assessment of safety related systems was performed in accordance with the development of general methodology for the NPP probabilistic safety analysis (PSA) in the USSR. The methods are based on the present-day advances in the field of NPP safety systems reliability and meet the main requirements placed on system analysis in performing the PSA. The methodical principles are implemented in SHARM-2 computer package used for the RBMK system reliability assessment. The main results of methodology and computer code development are given. (author). 3 refs, 1 fig., 3 tabs.

348
(IAEA-TECDOC–599, pp. 219-223)
Development of technical specification surveillance requirements for Sizewell "B" power station. Sargeant, W.B. (Nuclear Electric plc, Knutsford, Cheshire (UK)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

The paper describes the adaptation of Standard Technical Specifications to the licensing requirements of the United Kingdom for the first PWR to be built by Nuclear Electric plc (formerly a part of Central Electricity Generating Board). The application of probabilistic methods in the design and safety analysis is described, and the decisions to be taken on the scope, structure and interdependence of the technical specifications for Sizewell "B" Power Station are assessed. (author). 1 tab.

349
(IAEA-TECDOC–599, pp. 225-233)
Control of power dependent safety margins. Haeusermann, R. (Kernkraftwerk Leibstadt AG, Leibstadt (Switzerland)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

The Leibstadt Plant KKL, situated on the river Rhein in Switzerland, is a BWR-6, Mark III, GE-Plant with a BBC turbogenerator set with a net electrical output of 990 MW. In December 1973, the work contract was signed and in December 1984 KKL started the commercial operation. The construction period was influenced by the TMI-incident in 1979. The incident prompted a very tedious design review by the authorities, vendor, and KKL. The PRA-studies were part of the design review. (author). 1 ref., 7 figs, 1 tab.

350
(IAEA-TECDOC–599, pp. 245-251)
Use of PSA to evaluate operating strategy compliance with operating policies and principles requirements. Dick, B.N.; Lawrence, P.N. (Ontario Hydro, Toronto, ON (Canada)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

Within the Canadian regulatory environment, the Operating Policies and Principles (OP and Ps) define the operational safety philosophy and the limits and conditions for safe operation of our nuclear generating stations. As is the case with Technical Specifications, these limits and conditions are, for the most part, based on deterministic safety analysis and engineering judgement. However, unlike the Technical Specifications, the OP and Ps are intended to specify a minimum number of key constraints, and to specify them in very broad, general terms. A program to perform level 3 Probabilistic Risk Assessments (PRAs) at all of Ontario Hydro's nuclear generating stations is being undertaken by the corporation's design organization. At present, the PRA for one station, Darlington-A nuclear Generating Station, has been completed with work underway, or planned, for the remaining nuclear generating stations. This paper describes several examples to illustrate how the Darlington PRA has been used to evaluate proposed operating strategy compliance with the requirements embodied in the broad, conceptual limits defined in the OP and Ps. The paper concludes with a discussion of planned and potential future developments, including the more extensive use of PRAs in the day-to-day operation of Darlington Nuclear Generating Station. (author). 3 refs, 4 tabs.

351
(IAEA-TECDOC–599, pp. 271-278)
Evaluation of WWER 440 technical specifications using PSA. Kovacs, Z. (Research Inst. of Fuel and Energy Complex, Bratislava (Czechoslovakia)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

In the paper two case examples are chosen to demonstrate revision of WWER 440 technical specifications regarding surveillance frequencies and out-of-service times. Two V-213 type units have the same Reactor Protection Systems (RPS), but different test intervals for measuring channels, namely: (a) each channel has to be demonstrated operable once each month; (b) each channel has to be demonstrated operable once every two months. In case of the second case example, allowed outage time AOT risk measures at the system level were calculated for the components of High Pressure Core Cooling System (V230 type reactor). (author). 5 refs, 5 figs, 1 tab.

352
(IAEA-TECDOC–599, pp. 279-287)
Risk based operating configuration management. Schmidt, E.R.; Fulford, P.J. (NUS Corp., Gaithersburg, MD (USA)). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

This paper discusses the development of methods and software as well as experience with the utilization of a PRA for configuration management related activities such as: Planning maintenance activities; Checking, confirming, and justifying allowable outage times and, in the long-term, possibly replacing the license technical specifications; Providing information for development of accident management activities. The impact of components being out of service for test and maintenance is to increase the core damage frequency (CDF) over that with all components nominally available. This new CDF represents the "instantaneous" CDF for the current configuration. A number of difficulties arise in providing a technically valid assessment of the current configuration risk in short turn-around times. These include: Completeness of modeling; Conservative treatment of unlikely (non-important) scenarios; Ease of updating model for design changes; Model size. Two basis approaches are available to provide the configuration dependent assessment of CDF: Modifying, and then re-solving the event tree/fault tree models to reflect the actual configuration; Modifying and requantifying a core damage Boolean equation to reflect the actual configuration. A computer program designed around the second approach has been developed. This program, NURISK, allows systems, trains, or components to be taken out of service or restored to service by specification of system, component type or specific component identifying number. Initiator frequency can also be changed to account for such things as switchyard maintenance or RPS testing. (author). 4 figs, 2 tabs.

353
(IAEA-TECDOC–599, pp. 235-244)
Risk-based evaluation of technical specifications for a decay heat removal system of an LMFBR plant. Hioki, K.; Kani, Y. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Apr 1991. 293p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92603285. (CONF-9006379–).

From Technical committee meeting on the use of probabilistic safety assessment to evaluate nuclear power plant technical specifications; Vienna (Austria) (16-22 Jun 1990).

In Use of probabilistic safety assessment to evaluate nuclear power plant technical specifications.

PNC has been performing Probabilistic Safety Assessment on the prototype fast breeder reactor Monju since 1982. Objective is to construct a probabilistic model for Monju for the overall safety assessment. This paper presents a method of applying probabilistic technique to the development of the Technical Specifications for the Decay Heat Removal System (DHRS) of a LMFBR taking into consideration both the outage risk and shutdown risk. The DHRS is usually redundant and stands by while the reactor is in power operation. Therefore, partial failure of DHRS can be repaired without shutting down the reactor. However, the reliability of DHRS is lowered due to the repair outage. And the probability of occurrence of initiating events that require a plant shutdown and DHRS operation increases as the repair is continued. On the other hand, if the reactor is shut down manually after detecting the failure, the operation of DHRS whose reliability is deteriorated is needed. From this point of view, a manual shut down can be considered as one of initiating events. Most of the Technical Specifications have been developed based on deterministic methods or engineering judgements. However, for a new type of reactor such as LMFBR, they should be determined based not only on the experiences of LWRs but on a new concept of risk because of the difference of plant design and characteristics. The basic concept of the method is to minimize the total risk or to keep it less than a preset limit. The method can be used to help plant operator decide if the plant should be shut down or not and, if the operation should be continued, help him determined the allowable outage time (AOT) and the test interval of remaining intact loops. We expect that this method will be combined with the "living PSA" and construct and on-the-site system which calculates the plant risk level in the real time mode and gives the AOT and test intervals which are best for the plant safety. (author). 3 refs, 8 figs.

354
(IAEA-TECDOC–612)
Analysis of replies to an IAEA questionnaire on procedures for accreditation of training programmes and for authorization and licensing of nuclear power plant operations personnel. (International Atomic Energy Agency, Vienna (Austria)). Jun 1991. 21p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605448.

The intent of this survey was to gather data and present results to facilitate the international exchange of information and experience in this field. This was accomplished using a questionnaire developed by the IAEA that was sent in September, 1989 to the thirty-one Member States having nuclear power plants operating or under construction. Of these, twenty-one responses were received. The questionnaire was constructed in two parts: (1) Accreditation of Training Programmes; and (2) Authorization and (Formal) Licensing of Operations Personnel. The analysis was conducted by IAEA staff with the assistance of consultants and resulted in an identification of the commonalities and differences in approach on these issues. An advisory group subsequently considered this analysis, interpreted the results relative to the original responses, and reached conclusions that are contained in this document.

355
(IAEA-TECDOC–615)
Nuclear applications for steam and

hot water supply. (International Atomic Energy Agency, Vienna (Austria)). Jul 1991. 122p. OSTI; NTIS (US Sales Only); INIS. Order Number DE92605449.

An increase in the heat energy needs underlined by the potential increase in fossil fuel prices, particularly in oil supplies, and by the necessity for an improvement of the environment worldwide, as signalized by the IAEA Member States, prompted the decision to start a programme leading to this report. This document is intended to help to identify the experience of Member States where nuclear power plants or specialized nuclear heat plants are employed or envisaged to be used for distribution of steam or hot water to industrial or residential consumers, covering low and medium temperature ranges. 25 refs, 33 figs, 15 tabs.

356

Operating experience with nuclear power stations in Member States in 1990. Vienna (Austria); IAEA (1991). 856p.

This report is the twenty-second in the Agency's series of annual reports on operating experience with nuclear power stations in Member States. As in previous years, in addition to annual performance data and outage information, the report contains a historical summary of performance and outages during the lifetime of individual plants and four figures illustrating worldwide performance and statistical data. It is hoped that this report and related Agency publications will be useful to everyone concerned with nuclear power reactors.

357

Analysis of load factors by vendors. Knox, R. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 15-16 (Jun 1991).

A table of power reactor load factors for the year 1990 was published in the April issue of Nuclear Engineering International. This article analyses those figures to investigate the performance of different vendors' nuclear units during 1990. (author).

358

The Indian experience - a utility perspective. Kati, S.L. *Nuclear Engineering International (Incorporates Nuclear Power) (United Kingdom);* 36: No. 443, 38-39 (Jun 1991).

India's total installed nuclear capacity is projected to be about 164 000MWe by the turn of the century. The current fuel mix as well as that projected by the year 2000, is tabulated. Lessons learnt from 70 reactor years of experience with nuclear power in India, and their effect on plans for the future are considered. (author).

359

Robots in nuclear power plants. Jurisica, L.; Zachar, P. *Jaderna Energie (Czechoslovakia);* 37: No. 7, 266-269 (Jul 1991). (In Slovak).

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia, at USD 10.- per typewritten page.

The fields of application of robots in nuclear power plants and the technical requirements placed on their design are characterized. The designs of robots developed in various countries over the world are briefly described. (Z.M.). 23 refs.

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