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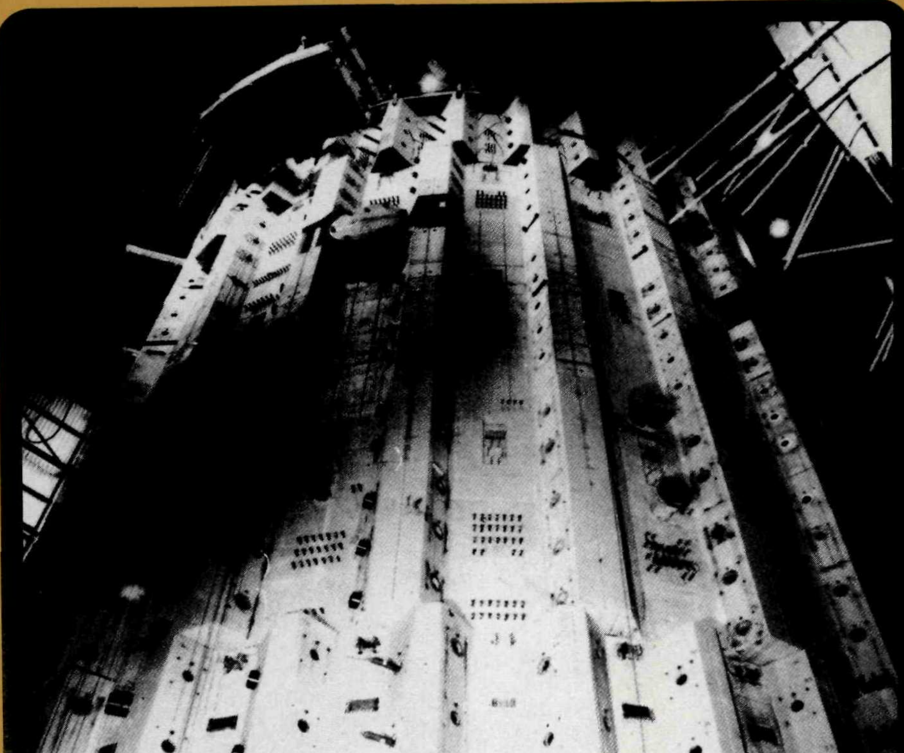

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The Nuclear Operations Analysis Center (NOAC) is pleased to announce the availability of *Status of Conversion of NE Standards to National Consensus Standards,* an annual publication prepared by the Performance Assurance Project Office at the Oak Ridge National Laboratory for the Department of Energy. The report provides information on the status of efforts to transfer the Department's nuclear energy program experience to the private sector for commercial use through the adoption of requirements from DOE Nuclear Energy (NE) Standards in National Consensus (e.g., non-Government) Standards. This report is available to DOE and DOE contractors from the Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831, FTS 626-8401. The report is also available to the public from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Rd., Springfield, VA 22161.

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STATUS OF CONVERSION OF NE STANDARDS TO NATIONAL CONSENSUS STANDARDS

Performance Assurance Project Office

S. D. Jennings

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The Nuclear Operations Analysis Center

NOAC performs analysis tasks, as well as information gathering activities, for the Nuclear Regulatory Commission.

NOAC activities involve many aspects of nuclear power reactor operations and safety.

NOAC was established in 1981 to reflect the broadening and refocusing of the scope and activities of its predecessor, the Nuclear Safety Information Center (NSIC). It conducts a number of tasks related to the analysis of nuclear power experience, including an annual operation summary for U.S. power reactors, generic case studies, plant operating assessments, and risk assessments.

NOAC has developed and designed a number of major data bases which it operates and maintains for the Nuclear Regulatory Commission. These data bases collect diverse types of information on nuclear power reactors from the construction phase through routine and off-normal operation. These data bases make extensive use of reactor-operator-submitted reports, such as the Licensee Event Reports (LERs).

NOAC also publishes staff studies and bibliographies, disseminates monthly nuclear power plant operating event reports, and cooperates in the preparation of *Nuclear Safety.* Direct all inquiries to NOAC, P.O. Box 2009, Oak Ridge National Laboratory, Oak Ridge, TN 37831-8065. Telephone (615) 574-0393 (FTS: 624-0393).

Cover: Our cover picture this issue shows the prestressed concrete pressure vessel for the THTR in the course of construction. The THTR was built near the village of Hamm in Westphalia, Germany, by a European consortium. The role of such vessels in the containments of gas-cooled reactors is discussed in this issue in an article "Containments for Gas-Cooled Power Reactors: History and Status" (photo used with the kind permission of Hochtemperatur-Kernkraftwerk GmbH, Gemeinsames Europäisches Unternehmen, Siegenbeckstrasse 10, D-4700 Hamm 1, Westfalen).

Nuclear Safety is a review journal that covers significant developments in the field of nuclear safety

Its scope includes the analysis and control of hazards associated with nuclear energy, operations involving fissionable materials, and the products of nuclear fission and their effects on the environment

Primary emphasis is on safety in reactor design, construction and operation, however, the safety aspects of the entire fuel cycle, including fuel fabrication, spent fuel processing, nuclear waste disposal, handling of radioisotopes and environmental effects of these operations, are also treated

Qualified authors are invited to submit articles, manuscripts undergo peer review for accuracy, pertinence, and completeness Revisions or additions may be proposed on the basis of the results of the review process Articles should aim at 20 double-spaced typed pages (including figures, tables and references) Send inquiries or 3 copies of manuscripts (with the draftsman s original line drawings plus 2 copies and with black-and white glossy prints of photographs plus 2 copies) to E G Silver, Oak Ridge National Laboratory, P O Box 2009, Oak Ridge, TN 37831-8065

The material carried in *Nuclear Safety* is prepared at the Oak Ridge National Laboratory's Nuclear Operations Analysis Center, which is responsible for the contents *Nuclear Safety* is funded by the U S Department of Energy (DOE), Office of Nuclear Energy, Office of Nuclear Safety Policy and Standards, and by the U S Nuclear Regulatory Commission, Office of Nuclear Regulatory Research Editing composition, makeup, and printing functions are performed by the DOE Office of Scientific and Technical Information (OSTI) Sale and distribution are by the U S Government Printing Office, see the back cover for information on subscriptions, postage, and remittance

General Safety Considerations

Edited by G. T. Mays

Report on the International Symposium on the Use of Probabilistic Safety Assessment for Operational Safety—PSA '91

By S. Chakraborty[a] and M. Khatib-Rahbar[b]

Abstract: *The International Symposium on the Use of Probabilistic Safety Assessment for Operational Safety, also known as PSA'91, was held at the International Atomic Energy Agency (IAEA) in Vienna, Austria, on June 3–7, 1991. During the symposium, 61 papers and 27 posters were presented in 13 sessions on subjects including PSA methodology, PSA use and applications, aging, common-cause events, human factors, living PSA, PSA Levels 2 and 3, and applications for process facilities. This article provides a brief overview of each paper presented at PSA'91.*

Publication of the landmark *Reactor Safety Study* (WASH-1400) in the early 1970s set the stage for probabilistic assessment of severe accident risk resulting from operation of nuclear power reactors. The past 20 years have seen substantial advancement in Probabilistic Safety Assessment (PSA) methodology, use, and application in operational safety, regulatory decision making, and plant design. Refinement of methods and increased computational capabilities have made the performance of PSAs more manageable and the results more valuable. As PSAs have gained wider recognition and acceptance globally, regulatory organizations, operating utilities, and reactor vendors have made substantial progress in implementing and using the PSA process and results for a variety of safety, operation, and design issues.

This progress motivated the American Nuclear Society (ANS) and the European Nuclear Society (ENS), together with various cosponsoring organizations, to organize PSA conferences every two years, alternating in the United States and Europe. The last meeting was held in 1989 in Pittsburgh, Pa. In 1991, IAEA organized this symposium as PSA '91, with cosponsorship from ANS, ENS, and the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). The purpose of the symposium was to present practical experience in the use of PSA for operational safety, with special emphasis on the state of the art in improving safety and the lessons learned on the basis of performance and application of past PSAs.

The symposium included nearly 300 participants from 32 countries (including 14 developing countries) plus 4 international organizations. During the symposium, 61 papers and 27 posters were presented in 13 sessions on the following subjects:

- Session 1: Invited Papers
- Session 2A: Methodology
- Session 2B: Regulatory Applications
- Session 3A: PSA Use and Applications (Part I)
- Session 3B: Applications for Process Facilities and International Activities
- Session 4A: Operating Experience and Aging
- Session 4B: PSA Use and Applications (Part II)
- Session 5A: PSA Results and Insights (Part I)
- Session 5B: Common Cause and External Events
- Session 6A: PSA Results and Insights (Part II)
- Session 6B: Human Factors
- Session 7: Living PSA
- Session 8: PSA Level 2 and 3

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In addition, a panel debated the "Research and Development Needs for Completeness and Credibility of PSA." For the promotion of wide dissemination of new PSA techniques and computer codes, throughout the symposium a special room with 12 personal computers was set up to demonstrate PSA-oriented computer programs. Fifteen different companies supplied software products for demonstration.

SESSION 1

INVITED PAPERS

Two invited speakers described the historical developments and regulatory implementation of state-of-the-art PSA techniques. Mr. L. Carlsson of Sweden explained how the PSA is used as a component of the periodic safety reassessment of Swedish nuclear power plants. He summarized Swedish experience from the 1980s and discussed the prospects for the 1990s. Mr. R. Caro from Spain then gave an overview of the regulatory and licensing uses of PSA in Spain.

SESSION 2A

METHODOLOGY

This session comprised six papers presented on topics including human reliability, dynamic PSA, Markovian analysis, benchmark exercise, safety monitoring and tracking, and safety criteria.

The first presentation by Dr. Mosleh of the University of Maryland was focused on a new approach to analysis of human actions in postaccident conditions and introduced ways of modeling errors of commission through the use of dynamic probabilistic risk assessment (PRA) methodology. This presentation sparked such questions as (1) a clarification of the benefits and limitations of dynamic methodology and (2) the various forms of operator error that can be simulated.

Mr. Arien from Belgium presented the second paper of the session, encompassing Markovian modeling of large systems and the development of a computer code for real applications. Feasibility and practicality of this method, as well as its potential limitations, were discussed following this presentation.

The third paper, presented by Mr. Abe of Japan, reported on a Japanese benchmark exercise on fault tree analysis. Results of this exercise showed significant variability among the various team results, both in their estimate of overall system reliability and the ranking of the most important contributors. Questions on this presentation focused on boundary conditions, ground rules for the benchmark study, and interpretation of the results.

Mr. Grozovskij of the USSR presented the fourth paper. He reported on an approach for monitoring trends in "safety signals" emitted from plants when viewed as integrated man–machine entities. Spectral analyses of the reports and incidents, production fluctuation, and other indicators were evaluated to show the variations in time. The discussions on this paper considered the issue of time delay from event occurrence to the reporting of the incidents and the prediction power of the methodology.

The fifth paper was presented by Dr. Lederman from IAEA. It discussed the Agency efforts to establish an internationally acceptable definition of risk and numerical safety goal. The paper generated many questions regarding the basis for establishing numerical criteria, their interpretation, and use.

A high level of operational safety can be achieved by implementing a coherent set of criteria, standards, and practices related to power-plant operation. The framework of the Probabilistic Safety Criteria (PSC) proposed in Dr. Lederman's paper embraces three regions of risk: (1) an upper region in which the risk is judged to be too high to make the practice or activity intolerable no matter what its benefit is, (2) an intermediate region in which the acceptance of the risk is subject to the overriding requirement that all reasonable practical measures have been taken to reduce the risk, and (3) a lower region in which the risk is judged to be low enough to be accepted without additional effort to further reduce it. The benefits provided by such a framework include an immediate indication of what is clearly acceptable or unacceptable, an effective means to direct both designer and regulatory efforts to areas in which further resources could be allocated to achieve further risk reduction, and an avoidance of viewing PSC as absolute go/no-go rules and the associated "number crunching syndrome." PSC is a quantitative expression of the desired level of safety, which guides the practices related to the operation of the plant, including the consideration of accidents in the training of personnel; the preparation of accident management procedures; and the provision of necessary organizations, services, and equipment to respond to accidents, should they occur.

The last paper was presented by Dr. Siu of the Massachusetts Institute of Technology who discussed event trees and dynamic event trees. The paper applied the dynamic event-tree approach to analyze steam generator tube rupture events, in response to the stated theoretical weaknesses in the traditional event/fault tree methods Simulation results were compared with event tree methods, showing that dynamic analysis can lead to numerically different accident sequence frequencies. Ways of generating operator errors were also discussed in this presentation. Questions included the range of applicability and practicality of the methodology and the role it can play to enhance the current nondynamic approaches.

The current PRA methodology for accident sequence analysis lacks proper integration of the plant system logic models and human actions. Traditionally, the human reliability analysis is performed in an ad hoc manner wherein the plant system logic models are developed first and then the impact of key human actions (identified on the basis of expert judgment) is added onto the detailed system models for the overall risk evaluation. This approach tends to undermine the real impact of human actions The use of a Markovian (memoryless) framework for accident sequences is challenged by the observation that the operators have memory, their beliefs at any given point in time are influenced (to some degree) by the past sequence of events and by their earlier trains of thought.

For the treatment of the dynamic interplay between operators and the plant, a modeling framework for dynamic accident sequence analysis needs to carry information on current hardware status, current levels of process variables, current operator "state of mind," scenario history, and time The dynamic event tree approach allows for event sequence branching at discrete points in time and explicitly incorporates information needed to provide the context for dynamic operator actions. The branching space is determined by a number of factors, including the possible plant hardware states, the operator's understanding of the plant condition, the set of actions planned by the operators (defined largely in terms of the operating procedures being followed), and the operating crew's current state At the end of every time step, system states are evaluated by using failure data and used as boundary conditions for the next time step However, the dynamic event tree approach requires a very large computer memory capability to keep track of accident sequence expansion.

SESSION 2B

REGULATORY APPLICATIONS

The six papers presented in this section addressed the increasing role of probabilistic studies in the regulatory decision-making and licensing process Together with the traditional deterministic safety analyses, PSA is becoming an effective vehicle for regulatory authorities to review the adequacy of design bases in meeting public health and safety requirements.

Dr. Schmocker of Switzerland provided an overview of the approach being used by the Swiss Federal Nuclear Inspectorate (HSK) to review the Swiss plant-specific PSAs. Regulators there have been using PSA methods for regulatory decision making since 1977 and in the mid-1980s required the performance of full-scope Level 1 and 2 studies (including external events) for all Swiss nuclear power plants At the end of 1990, extension of these studies was required to include the startup, shutdown, and outage phases A complete reassessment of PSA studies using alternative methods forms the basis for the Swiss review process This approach provides the licensing authority the bases for an independent confirmation of the significant PSA findings, conclusions, and important insights regarding plant safety features in mitigating severe accident vulnerabilities. The Swiss paper identified lessons learned from an in-depth review of the Muhleberg PSA. In particular, it demonstrated the potential pitfalls of eliminating major risk vulnerabilities by introducing a probabilistic cut-off criterion based on Level-1 PSA results.

The paper presented by Dr. Virolainen of Finland surveyed the Finnish methodology as applied to Loviisa (Units 1 and 2) and TVO (Units 1 and 2) Level 1 PSA studies. It described the aims, objectives, and findings of regulatory reviews by STUK as well as changes and modifications made in plant hardware and operational procedures implemented as a result of PSA findings. The paper also outlined the implications of PSA reviews to regulatory decision making.

The review of the TVO PSA noted the complexity of the TVO electrical power supply system. It addressed dependencies between a-c and d-c systems that decrease the reliability of the power supply system in general. The review also addressed the design originating diesel generator dependency on three distinct d-c buses, the nonconservative success criterion of the safety relief system, and the need for modifications in some emergency operating procedures. The review of the Loviisa PSA addressed conservative assumptions used in the analysis and several design deficiencies in plant systems that made a significant contribution to core-melt frequency. The utility had to make prompt plant changes to improve PSA-identified deficiencies. The paper also summarized major modeling deficiencies and the methodological strength of the Loviisa PSA

The Canadian paper, presented by Mr. Wild, discussed "Aspects of the AECB PSA Validation Program."

The role of PSA in Canadian reactor regulation is currently undefined. From the review of the Darlington Probabilistic Safety Evaluation (DPSE), the Canadian Atomic Energy Control Board (AECB) has started a PSA validation program. The principal objective of the AECB PSA Validation Program is the systematic identification and resolution of issues associated with the use of PSA in decision making. The program aims to provide the licensing authority with insights into PSA findings without going through a detailed reassessment of the whole study. The AECB is attempting to develop an analytical framework for evaluating the products of a PSA without having to evaluate the process. The program is still in progress, and the merit of the approach is uncertain at this time.

The paper from South Africa, presented by Dr. Kussman, gave an overview of the use of probabilistic studies in regulatory decision making in the Republic of South Africa. A computerized PRA methodology for use on personal computers is currently being developed by the regulatory authority in cooperation with the utility to assist in the assessment of licensing issues and to perform proactive safety work. The methodology is similar to the recently published NUREG-1150 (Severe Accident Risk) study in the United States. A plant-specific data base composed of about 900 incident reports and a Bayesian approach were used in the data analysis. Human cognitive psychology was used in the human error analysis, and the Beta-factor method was used in the common-cause failure (CCF) analysis. PSA is included as part of the initial license submittal for nuclear facilities for regulatory purposes as well as to assist in decision making on licensing issues.

Mr. Perez discussed the effort under way in Mexico for performing a PSA for the Laguna Verde Nuclear Power Plant [a GE boiling-water reactor (BWR)/5 Mark II design]. For the safety review of the plant, the Mexican regulatory authorities rely not only on the classical deterministic approach but also on the results and insights of plant-specific PSAs. The paper discussed the detailed framework of methods and computer codes being used for the review of the Level 1 PSA study for Laguna Verde. In the future, the PSA methodology will also be applied to arrive at the bases for, and evaluate the adequacy of, emergency operating procedures and Technical Specifications. It is also foreseen that a further application of PSAs could entail the development of a set of probabilistic safety criteria.

Mr. Koca from Turkey outlined the preparedness of the Turkish Atomic Energy Authority on the licensing and regulatory work that would be required if a nuclear power plant were built in Turkey.

SESSION 3A

PSA USE AND APPLICATIONS (PART I)

Dr. Toth from the PAKS nuclear power plant in Hungary presented the first paper in this session. He described an application of the available plant PSA to improve allowed outage time (AOT) restrictions. Current practices call for limited AOTs with concurrent testing of the other redundant systems.

The PSA is being used to assess the core damage frequency compared with a base case (i.e., normal operations with a six-week test interval). The code PSAPACK (developed by the IAEA) is being used, and the comparison has been completed for the diesel generator AOT. The most important preliminary finding is that the existing 24-hour AOT with testing of the other trains is very conservative and that an AOT of 4 to 8 hours might be adopted without testing of redundant trains.

The second paper was presented by Dr. Holloway (U.K.) on behalf of a joint German–British team. Its subject was the application of the PSA to accident management development for the 23-MW research reactor FRJ-2 at Julich. The design of the reactor is such as to cause a potential vulnerability to loss-of-coolant accidents (LOCAs). At least four different LOCAs with different conditions (especially success criteria) have been assessed. After screening of the sequences with generic system failure rates, the risk dominant accidents were analyzed in more detail. In three examples, rather simple but effective accident management modifications improved the defense in depth. It was noted that the logical structure of the PSA allowed the needs to be seen clearly and that uncertainties were thus not important.

Dr. Holmberg from Finland discussed uncertainty and sensitivity studies supporting the interpretation of the results of the PSA for TVO Units 1 and 2. Both sensitivity studies of a straightforward nature and uncertainty statistical studies were made. The uncertainty analyses were performed on the most important (top 100) cut sets with a Monte Carlo technique, with separate studies for the propagation of modeling uncertainties. It was found that uncertainties concerning operator actions could affect the results noticeably. The uncertainty study was found to be an effective internal review process for the PSA.

The second part of this session focused on more generic applications of PSA. The use of plant-specific PSA based on operational data (component and human reliability) to detect and correct operational weakness is beneficial for different activities related to plant day-to-day operation. One of the areas in which plant-specific PSA has been used is the analysis of operational events.

An approach to decide the safety significance/safety importance of individual reported events was the topic of the paper presented by Mr Tolstykh of the IAEA This approach includes (1) understanding the incident and its safety implications by a knowledge of plant design, operation, and the contents of specific PSA, (2) relating the incident to the PSA models by determining which accident sequences are involved or could be involved, which fault trees and basic events are of concern, and what recovery actions could be applied, (3) modifying the models to reflect the incident by restoring accident sequences that originally truncated out of the final results, changing basic event probabilities, and evaluating new human error rates, (4) recalculating sequence frequencies, regenerating system and sequence minimal cut sets when needed, and calculating several importance measures, and (5) drawing insights by comparing the conditional core damage probability with the overall core damage frequency and deciding the new dominant contributors to the core damage frequency and the new importance of remaining systems–components–operator actions to prevent core damage Case studies were presented in which events analyzed in the Accident Sequence Precursor Study were reassessed in a plant-specific context The IAEA Incident Reporting System (IRS) now contains over 1000 reported events, many of which could be assessed by this technique

Dr Fulford of the NUS Corporation (U S A) described further developments in the NUS risk methodology as applied to Individual Plant Evaluation (IPE) studies NUS has developed NUPRA, a Level 1 PSA code, and the paper described the related code NUCAP+, a tool for containment event tree analyses (an important element of Level 2 PSAs) NUCAP+ is a powerful tool allowing for decomposition of event trees into subtrees so that they can be kept to a manageable size

Dr Ellia-Hervy of the Framatome PSA department described the wide-ranging applications now being made in the French PSAs The PSAs for the 900-MW(e) and 1300-MW(e) plants were used to improve defense against identified vulnerabilities, particularly in the shutdown, midloop operation in which vulnerabilities to CCF have been identified The PSA insights are also being used to direct designs so that accident problems can be avoided or mitigated Thus, for example, additional automation can be included in cases in which the time available for human action is very short PSA insights are also leading to simplifications in operation and maintenance, which will improve plant availability as well as safety

SESSION 3B

APPLICATIONS FOR PROCESS FACILITIES AND INTERNATIONAL ACTIVITIES

The first paper, presented by Mr Haddad of IAEA, compared PSA applications and methodologies in nuclear and nonnuclear process industries Although the concepts are the same, the methodological practices, applications, and implementation are quite different The authors assumed the main reason for these differences is that the nuclear industry is basically a one-process industry, whereas the chemical process industry is characterized by a multitude of chemical processes As an example, the concept of a "living PSA" is only used in the nuclear industry, whereas the technique of Hazard and Operability (HAZOP) studies is only used in nonnuclear applications The authors concluded that an interchange of methods and applications would benefit both industries For example, the rigorous application of interdisciplinary brainstorming sessions in HAZOP studies would be beneficial in the process of conducting a PSA for the nuclear industry

Mr Turner of the United States described the methodology for risk assessment as applied in another part of the nuclear fuel cycle, specifically U S gaseous diffusion plants In such plants both nuclear and chemical hazards are present The chemical hazards are especially of concern because UF_6 as well as other chemicals are used in these plants The approach is basically the same as that used in other chemical industries

- Hazard identification by use of HAZOP studies
- Accident sequence development by applying event tree analyses
- Risk assessment

Fault tree analyses are used to estimate the frequencies of various accident scenarios, whereas accident consequences are estimated on the basis of plant operating experience and industry data aided by application of plume dispersion models

The third paper showed the necessity of applying risk assessment techniques in space programs An overview was given of the PSA techniques as developed by the European Space Agency In this application extensive use is made of expert judgment techniques

The next two papers provided overviews of the work being performed under the respective auspices of the Commission of the European Communities and the OECD/CSNI Principal Working Group 5 on Risk Assessment The last paper summarized the coordinated research program on reference studies on probabilistic modeling of accident sequences performed under the auspices of the IAEA.

SESSION 4A

OPERATING EXPERIENCE AND AGING

Some of the oldest plants have already experienced approximately 30 years of operation. Because aging can lead to loss of component function and impaired safety, aging of components has become a significant safety concern worldwide.

Mr. Vesely of the United States presented the first paper in this session. He discussed the effects of active component aging on core-melt frequency. This work is based on the linear aging rate model with postulated aging rates established by an expert judgment process. The paper showed that, with the use of the linear aging model in conjunction with an existing PRA, and modeling first- and second-order aging effects on component failure rates, core-melt frequency was postulated to increase by several orders of magnitude provided that no preventive measures are taken. However, it was noted that the high core-melt results would be significantly lower if component replacement strategies were properly factored in. Despite some shortcomings caused by data limitations, the prioritization of risk importance does allow the identification of components whose aging effects are most important.

Dr. Bier of the University of Wisconsin presented a paper discussing the issues associated with estimating linear aging rates. She indicated that the aim of data trending analysis is to quickly detect an adverse trend but not to make false calls in detecting bona fide increases in failure rates. She summarized the problems associated with performing actual data trending: (1) the failures are typically sparse and (2) maintenance and replacement strategies change over the life of a plant. The results of 17 years of data trending at one U.S. BWR were reviewed. Many components revealed no detectable aging trend, a few components exhibited some indications of increasing failure rates, whereas others indicated decreasing failure rates with age. This work indicated that the specific choice of the initially assumed failure rate is a crucial assumption when testing for an increasing or decreasing failure rate trend.

Most models for detecting aging-related trends in event frequencies are characterized by an initial frequency and a rate of increase in that frequency. The conclusions derived from these models are very sensitive to the choice of prior distributions. Moreover, in most practical situations the actual failure rate may be of more concern than its rate of increase. Bier's paper revealed that estimates of future event frequencies are much less sensitive to the choice of prior distributions than comparable estimates for the rate of increase. Fairly reliable means and upper bounds can be obtained for event frequencies several years into the future, even with extremely small numbers of data points. Thus it would be advisable to identify some threshold value of the failure rate that could serve as an appropriate trigger for corrective action and then estimate the probability that the failure rate will exceed that value within some specified time horizon.

Dr. Kurchsteiger of Austria presented a paper on computer software for monitoring safety performance indicators. It indicated that trending-type information is available at different levels. At the plant level, core-melt frequency can be trended on the basis of living PSA models. At the system level, system failures and downtimes for repairs can be trended as a system-level performance indicator. Similarly, individual component failures and downtimes can be trended as component-level performance indicators.

Mr. De Guio of France discussed EdF's data collection activities and their utilization in probabilistic studies. EdF is currently tracking about 500 different component types on each of its plants. Data are also being collected on plant power profiles, capacity factors, event occurrences, and human reliability.

Dr. Ilberg of Israel presented a paper concerning IAEA efforts to improve the definition of initiating events modeled in PSAs. The work will eventually lead to the issuance of an IAEA Guidebook showing appropriate methods to establish a complete list of initiating events and how to quantify their frequencies.

The next paper was given by Dr. Hirschberg of IAEA, focusing in the shifts occurring throughout the world on PSA modeling and usage. He indicated that PSA programs aimed at investigating the safety of existing nuclear plants are well under way. Most recent development activities have been addressing areas that amounted to major uncertainties in past work. Good examples include the treatment of the probability of human errors of commission and modeling of passive design features in the emerging Advanced Light-Water Reactors (ALWRs).

SESSION 4B

PSA USE AND APPLICATIONS (PART II)

Dr Kozuh of Yugoslavia considered the effect of high-pressure injection on the large-break LOCA event tree developed as part of the PSA for the Krsko Nuclear Power Plant [a two-loop Westinghouse pressurized-water reactor (PWR)] In the second paper, Mr Serbanescu of Romania presented a new approach to decision making in different phases of PSA studies, particularly in regard to system success criteria

Dr Novàkovà of The Czech and Slovak Federal Republic discussed the use of probabilistic methods to optimize Technical Specifications (Tech Specs) of VVER 440 reactors The current Tech Specs for VVER 440 reactors are based on deterministic analyses from Final Safety Analysis Reports complemented by engineering judgment Probabilistic assessment combined with operating experience data can be used as a valuable tool in revising Tech Specs, this will result in fewer plant shutdowns and increased plant availability Several examples were given of VVER 440 Tech Spec revisions regarding surveillance frequencies and allowed outage times

Dr Sato of Japan presented the use of PSA results in the design of a future generation of BWRs The new concept of severe accident (SA) tolerance design was introduced Several examples of SA-tolerable designs from their development program were provided Following that, Dr Gheorghe of the IAEA introduced the new idea of combining two seemingly different concepts to provide a vehicle to enhance operational safety

Dr Jaitly of Canada focused on the initiating event identification process and consequence analysis of the new generation CANDU 3 reactors, which are currently in the detailed design phase Including probabilistic safety studies as part of the design process was indicated as one valuable way of ensuring that all safety-related requirements are defined early This minimizes the probability of engineering rework during and after plant construction The activity of preparing a systematic review of initiating events (1) provides confidence that licensing and risk assessments of the plant design are well founded and (2) has the design benefit of identifying priority areas for design resolution or detailed consequence analysis

SESSION 5A

PSA RESULTS AND INSIGHTS (PART I)

Dr Berger of France presented an overview of the ongoing PSA activities at EdF Current plans are for the PSA models to be updated every 3 years with the use of equipment experience Specific PSA models have already been developed for the 900-MW(e) and 1300-MW(e) class PWRs Work is currently under way to develop PSA models for the 1500-MW(e) PWRs It was reported that shutdown modes were found to dominate the core-melt profile Several participants commented on the EdF assumption that the LOCA frequency at low-pressure shutdown conditions is the same as that at full power Dr Berger responded that this is acknowledged to be a major conservatism that will be eventually improved with further analysis

Dr Lanore of France presented a paper on low-power shutdown modes of risk of 900-MW(e) PWRs in France Significant operational data (200 years of reactor experience) were collected from 35 plants and averaged to establish the average annual power profiles of French PWRs These data were used in adjusting annual frequencies for various initiating events The results indicate that 60% of the core-melt frequency risk comes from full-power events, and the remaining 40% results from events occurring during shutdown mode The principal risk at shutdown results from the loss of cooldown capability caused by cavitation of the residual heat removal (RHR) pumps when the water level in the primary circuit is lowered Another important risk contributor, based on conservative assumptions, is the possibility of a reactivity accident caused by injecting diluted water into the core long after a shutdown

Dr Lanore also presented a paper on the results of probabilistic analysis of reactivity accidents in the 900-MW(e) PWRs The results indicate insignificant core-melt frequencies for control-rod ejection scenarios and steamline break scenarios with failure of shutdown control rods and boration Considerable work has been focused on accidents involving startup of an idle reactor-coolant-system (RCS) loop that was previously isolated In response to questions, Lanore acknowledged that considerable uncertainties exist in the physical modeling of such scenarios, in particular the core physics behavior and whether fuel-rod melting is, in fact, possible Because of inabilities to quickly resolve these uncertainties, an automatic system was developed to halt ongoing boron dilution events by tripping pumps. This improvement will lower the core-melt frequency to 10^{-6}/yr.

Mr. Moore of the United States discussed activities associated with the ongoing Borssele PSA in the Netherlands, which has been supported by a number of IAEA-sponsored technical support missions. Twenty-three internal events were analyzed in detail, and screening analyses were performed for fires and internal floods. Level 2 analysis was also performed with the STCP Code (for in-vessel accident progression) and the German multicompartment code WAVCO.

Dr. Bertrand of France presented a paper involving the use of mobile systems to increase the reliability of long-term heat removal following a nonisolatable rupture in the primary circuit. These mobile systems permit the low-pressure safety injection pumps to be backed up by a confinement spray system pump and the pumps of the latter system to be backed up by a mobile pump. These pumps are moved in, set up, and connected in the four days following an accident. With the use of the PSA model for the 900-MW(e) PWR, it was shown that for the long term (between 15 days and 1 year after an accident), the existence of mobile systems significantly reduces the probability of a core melt.

SESSION 5B

COMMON CAUSE AND EXTERNAL EVENTS

The first paper in this session was presented by Dr. Vaurio of Finland. He described a method for dealing with CCFs applicable to systems with up to four redundant trains. The method is based on actual observed CCF event data. Data are used to estimate multiple-failure occurrence rates directly rather than using ratios of multiple failures to single independent failures. This approach avoids certain inconsistencies present in current event data collection systems, such as underreporting of independent events. This method also accounts for differing test intervals, testing rules, and system success criteria. Numerical results were presented for many systems and component types.

The next paper, presented by Mr. Mankamo of Finland, compared the applicability of three CCF models to a highly redundant system, a BWR safety/relief system with 12 valves. It also described relationships between several fundamental variables of the problem.

Dr. Mosleh of the United States introduced a general taxonomy for the coupling mechanisms of CCFs, dividing them into hardware similarities, operation similarities, and location similarities. Application to pump failure data indicated typical differences between normally operating and standby systems. Generic defenses against the coupling mechanisms were also discussed.

Dr. Natta of France presented a model for non-simultaneous common-mode failures. The model is based on the observation in residual heat removal systems that the failure rate of the other redundant trains often increases whenever a failure is observed in one train.

The last paper in the session, presented by Dr. Norta of Finland, described a systematic fire risk assessment procedure, combining plant logic models developed for internal initiating events with the fire ignition and fuel properties of each room. In the screening phase, a fire was assumed to fail all components in the area, whereas a fire propagation and suppression analysis is carried out in the second phase for risk-critical areas.

In summary, an improvement in the field of CCF analysis requires a better understanding of failure causes, reasons for propagation of a failure event to multiple components, and the role of defense against CCFs. The investigation on coupling mechanism classification represents recent progress in the development of a systematic and practical approach to the assessment of plant-specific vulnerabilities to CCFs. However, the available methodologies for quantitative interpretation of the coupling mechanisms, root causes, propagation mechanisms, and defense are still rather weak. This indicates that, together with the qualitative analysis of coupling mechanisms, quantitative methods must be developed for a more realistic estimation of plant specific CCF frequencies. The endeavor to directly use the system-level CCF rates obtained from the reported CCFs suffers from the following observations: (1) the scarcity of data and the resultant uncertainty of failure rates and (2) not accounting for partial system failures (i.e., multiple failures that do not make the system unavailable).

SESSION 6A

PSA RESULTS AND INSIGHTS (PART II)

Mr. Bickel of the United States presented a paper discussing testing intervals for safety-related pumps and valves in the United States. Testing frequencies for these components are specified in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The required testing frequencies in the ASME Codes were originally established on the basis of engineering judgment before significant component reliability data (such as those used in PSAs) were available. PSAs tend to indicate that certain pairs of valves have very large risk importance. Some examples include motor-operated valves (MOVs) that isolate cooling water to the RHR heat exchangers and MOVs that isolate normal RCS makeup systems from high-pressure emergency core cooling systems. Current industry standards result in critical pumps being tested quarterly and critical MOVs being tested at cold shutdowns. Existing reliability data indicate that MOVs should be tested much more frequently than pumps. In response to questions from the audience, it was pointed out that the net risk implications have not yet been considered, only the unavailability of individual components.

Dr. Hirschberg of IAEA presented a paper on the IAEA-sponsored International Peer Review Service (IPERS) for IAEA member nations. He indicated that IPERS missions have already been conducted for Gorky (USSR), Borssele (Netherlands), Guangdong (P.R. China), Forsmark 1/2 (Sweden), and Cernavoda (Romania). Such peer review missions include document reviews, developing of a detailed set of questions, plant visits and meetings with the PSA analysts, preparing a draft report of the findings, and a formal discussion of results with the requesting organization. The stated objective of the overall IPERS program at IAEA is to bring international experience and guidance to improve a given PSA study (thus improving the results of safety-related applications based on the PSA). On the basis of the increasing number of requested IPERS missions, the program is fulfilling an important need. Requests for future IPERS missions were reported to currently exceed available IAEA budgets, but the agency is trying to accommodate the requests.

SESSION 6B

HUMAN FACTORS

Ms. Goktepe of Turkey presented a paper on the use of PSA insights for research reactor operator training. She presented the insights gained from the PSA of the 5-MW research reactor TR-2, especially as they relate to the human factors area. Investigation of human behavior under both normal and emergency conditions was emphasized. Event response sequence diagrams for the earthquake scenario were shown, and the methods of studying operator behavior under accident conditions were discussed.

The second paper was presented by Dr. Mosneron-Dupin of France and considered the lessons derived from the French experience regarding human reliability. It was indicated that the lessons were incorporated systematically in the PSA-1300 study of the Paluel Nuclear Power Plant.

SESSION 7

LIVING PSA

The first paper, presented by Mr. Ilberg of Israel, discussed an extensive review by IAEA of computer codes related to PSA Level 1. About 80 codes were covered and categorized into 11 classes on the basis of the particular field of application (e.g., fault tree analysis, uncertainty analysis). The information provided included descriptions of the codes and their attributes, the hardware required to run them, and the institute that may provide them.

The second paper, presented by Mr. Johanson of Sweden, described the Nordic project "Safety Evaluation, NKS/SIK-1," which involves the complementary development areas of living PSA and safety indicators. The main objectives of the project are to define and demonstrate the practical use of living PSA and operational safety indicators for safety evaluation and to identify possible improvements in operational safety. The project will also cover studies of problems related to risk decision making and to formulation of a suitable framework for use of PSA in safety-related decision making. Because work has just started on this project (and will continue for three years), the paper only discussed early parts of the work planned.

Mr. Lawrence of Canada described how the Darlington Probabilistic Safety Evaluation (DPSE) was used to develop an operational reliability program and the lessons learned during implementation of the program. The program used unavailability models based on detailed fault trees for 10 selected safety systems (1) to develop testing programs for the systems and (2) to provide input to the development of maintenance programs and operating procedures. During plant operation, these models are being used to review the impact of test program deviations, control system configuration, monitor system and component performance, and assess the potential impact of proposed design changes. An additional 20 safety-related systems were included in the program in a less rigorous manner. The generally positive experience with the operational application of PSAs will result in increased use of the technique for making operational decisions. Although the online use of a risk model is not imminent, the application of PSAs will likely be extended to the fields of training and the preparation of emergency operating procedures.

The next paper, presented by Mr. Matsuoka of Japan, discussed the development of a support system for the GO-FLOW reliability analysis to enable the GO-FLOW methodology to be widely used in a living PSA. The GO-FLOW methodology is a success-oriented system-analysis technique capable of evaluating system reliability and availability. Two principal steps in the GO-FLOW methodology are construction of a GO-FLOW chart and preparation of the input data for running the GO-FLOW analysis program on a mainframe computer. For large or complex systems, the effort required for these tasks can be substantial. The GO-FLOW support system was developed to aid in these tasks. This menu-driven analysis tool is fully integrated and based on a personal computer. This tool reduces the effort required to develop the GO-FLOW chart and automatically produces the input data for the mainframe computer.

Dr. Ancelin of France presented the experience gained, the difficulties met, and the developments achieved by EdF in constructing a completely computerized PSA (LESSEPS 1300) and in preparing this knowledge base for operational safety. The LESSEPS 1300 is a fully computerized knowledge base on the Paluel power plant, including methods, reliability models, and programs. At the end of 1991, the objective of this program is to point out different types of tools that could be dedicated to specific applications in the framework of operational safety.

The last paper in this session, presented by Dr. Kafka of Germany, concerned the use of PC-based PSA models. A principal conclusion from this paper pointed out that the last decade has seen a strong tendency to use PCs or workstations instead of mainframe computers in the field of PSA. However, a dependence on mainframe computers appears to remain for very large and intermeshed fault trees.

It appears that the most common living PSA application in the countries with nuclear power plants in operation is based on a continuation of already performed PSAs. Examples of this application were presented in a number of the previously discussed papers at this conference.

The Finnish Authority is now developing and implementing a living PSA program for its in-house PSA application. This program will result in a communication network of living PSA tools, modules, and data between authority and utilities.

The STARS (Software Tools for Advanced Reliability and Safety) project in CEC aims at providing a computer-based environment for supporting the PSA and accident management. The plant model for safety analysis has a hierarchical form representing systems, subsystems, and components. The generic part (application independent) of the plant model contains general behavior rules or models, general data, experience, and heuristics, which provide the intelligent support for the analyst to create the plant-specific part of the model. When performing a PSA, STARS has a set of reasoning modules supporting the creation of PSA models (fault trees, event trees) and the analysis of these models. The plant models can be modified easily by using engineer-oriented interfaces, such as CAD tools, for modifying system configuration. The impact of such modifications can be evaluated by the reasoning modules that develop PSA results from the plant description.

With the development of the living PSA concept, development of its supporting tools is also in progress. The following paragraphs give examples of such activity in several countries.

In Finland, the IBM PS/2-based STUK PSA (SPSA) codes are under development. In addition to graphical fault and event tree manipulation and quantification, a hypertext system includes the ability to expand on any basic event by calling up a variety of documentation sheets (including graphics) with a simple keystroke, eventually returning along the same path to the original screen.

In Japan, a PC-code network for a PSA was developed for a prototype liquid-metal fast-breeder reactor. It includes IEIQ (Initiating Event Identification and Quantification), MODESTY (Modular Event Description for a Variety of Systems), FAUST (Fault Summary Tables Generation Program), and ETAAS (Event Tree Analysis Systems). The QUEST code system includes the logical models and is intended for use by PSA analysts. The LIPSAS system includes only the cut set lists and a limited range of functions and is intended for use by non-PSA specialists at the plants.

In Sweden, the SUPER-NET code provides a data base and graphical editing tools for fault tree handling (input, editing, and modification) and quantification (sensitivity analysis, importance measures, and statistical uncertainties). The modules for time-dependent and life-cycle cost analysis are also included.

The U.S. IRRAS code includes graphical fault tree manipulation and various reduction–quantification options. Accident sequences are quantified by combining cut sets at the function level.

SESSION 8

PSA LEVELS 2 AND 3

The first paper in this session was presented by Mr. Hill of South Africa. It focused on the use of

Level 2 and 3 PSA for regulatory decision making. The paper discussed the need to link Level 3 PSA results to public risk criteria, which, in turn, leads to the need for a "living PSA" approach for rapid decision making. PSA has been used for regulatory decision making over many years, its use developing as methods have developed. The criteria used in South Africa were described, the methods outlined, and their use explained.

The next paper, presented by Dr. Harris of the United Kingdom, compared the methods for containment analysis in PSA. The objective of the study was to develop alternative methods of containment analysis that would be more efficient and more elegantly structured than the event tree approach currently used. Markov modeling and fault tree modeling were investigated as alternatives using a simplified problem of hydrogen deflagration. The Markov modeling led to a more compact representation of the containment analysis, with no intermediate binning required. The fault tree approach did not have the complexity of the event tree approach because it does not take explicit account of the evolution over time of the scenario. Despite this lack of time dependence, the fault tree analysis gives acceptable results. It was concluded that both methods are worthy of further investigation.

The last paper, focusing on the characterization of fission-product releases resulting from severe reactor accidents, was presented by Dr. Khatib-Rahbar of the United States. This paper provided a state-of-the-art review of the history of severe accident source term and methods for predicting radiological releases for both PWRs and BWRs. He indicated that the traditional, less-mechanistic, conservative, and often contradictory approach to fission-product source term analysis is slowly being replaced by a physically based, more consistent, analysis framework. The paper discussed several computer codes that have been developed in the United States, Europe, and Japan. The various phenomenological processes governing the evolution of fission-product release and transport were also discussed, and a detailed comparison of several computer codes was made. Potential areas of source term uncertainties were delineated, and the modeling deficiencies of the existing severe accident and source term codes were also outlined.

OBSERVATIONS

The PSA offers a powerful approach for use in improving design, operations, policy implementation, and regulatory decision making. Topics covered in PSA'91 included PSA usage in life extension studies, maintenance planning, determining allowed outage time, optimizing Technical Specifications, and plant licensing.

The human reliability analysis (HRA) research aims to provide guidance to practitioners in deriving human error rates for specific actions under accident conditions. However, such an engineering approach lacks an appreciation of the internal mechanisms (e.g., reasoning, association, and memory) that govern human behavior. The available data, mostly from simple routine activities performed by individuals, lack the important and key element of human behavior of a nuclear plant crew related to detection, diagnosis, and decision making following an accident. Because of these, HRA methods rely heavily on expert judgment to estimate human error probability. Moreover, as the result of the complexity of analysis, the treatment of dependencies between multiple human actions has mostly been ignored in HRA.

Very few papers presented topics included in Level 2 PSA. Severe accident phenomenology, containment performance, fission-product release, and accident and consequence modeling received minimal attention in this symposium.

PSA '91 provided an international forum for presenting papers concerning methods development and application. The meeting demonstrated that PSA usage in operational safety is receiving ample acceptance. The PSA-based safety analysis techniques are also proliferating rapidly throughout the international nuclear and process safety community.

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Good Relationships Are Pivotal in Nuclear Data Bases

By A. S. Heger[a] and B. V. Koen[b]

Abstract: *In a previous letter to the Editor [*Nuclear Safety, *Vol. 31, No. 4 (October–December 1990)], we expounded the importance of effective use of information in the nuclear industry. As a result, we received several requests for a full article on this subject. This article will start with segments of that letter; its tenet is that valuable information is stored in our nuclear experience data bases that must be capitalized on for enhanced operation of our plants, training, and rule-makings. After an introduction, a method of adaptive information retrieval based on neural network methodology is introduced and followed by an example.*

The Information Revolution has aggressively championed information technology as the key to effective operation and competitiveness. Information is said to be a "strategic asset," with which we agree.[1] In the nuclear power industry, the increased interest in improved plant safety, performance, and mandated probabilistic risk assessments has demonstrated the need for quality data in the form of information. The Holy Grail of information technology advocates is the seamless integration of hardware, software, and telecommunications technology into networks where engineers and risk analysts can get whatever information they need whenever they need it. We agree that this is a necessary, but not sufficient, component of information technology.

Experience has shown that this approach can be very confusing. This web of technological wonders has led to the inundation of end users to the point that they refuse to access the system. These users can be plant engineers, regulators, and other decision makers. These are the people who make important decisions that might affect the safety of a large segment of the society.

Realistically, information is not just a compilation of data that is based on experience. The molding of these data into meaningful "relationships" creates knowledge. Conventionally, this information has been formed by a formal and an informal network of colleagues who exchange their experience and knowledge. This has been particularly true in the nuclear power industry. For example, the American Nuclear Society (ANS), the Institute of Nuclear Power Operations (INPO), and the Nuclear Regulatory Commission (NRC) through different programs have advocated this mode of information development and, in some areas, have been very successful. With the advent of large central data bases, a new dimension has been added to this relationship: the interface between the computer and the user. So far the computer has played the role of a powerful but dumb partner in this relationship. This is one of the reasons why the human side of this partnership is frustrated. For the alleviation of this problem, the computer must acquire some form of human-like behavior: it must adapt to its user by learning the characteristics of its human counterpart. Although data are important, it is the information—the meaning imbedded in the data and the interaction between human and machine—that actually governs how effectively we operate a plant or manage an organization.

PROBLEMS ASSOCIATED WITH LARGE CENTRALIZED DATA BASES

Although it has become easier to retrieve data from large data bases, it often has become difficult to find the true meaning imbedded in these data. Because of the large sizes of these data bases, it is often impossible for the end user to "see" what actually is in those data bases and how to link the related pieces of data. The sizes of the central data bases keep growing, and the quality of the data flowing through the computer networks increases. In most power-generation organizations, the result is that technology is bestowing better quality data but a declining quality of information.

Effectively, the Information Revolutionists have been looking through the wrong end of the telescope.[1] For example, they enjoy pointing to the NRC Sequence Coding and Search System (SCSS) as a model of successful design; this is true undoubtedly because NRC's SCSS does a tremendous job of tracking and coordinating hundreds of thousands of pieces of Licensee Event

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Report (LER) data every day.[2,3] It is a superb example of an effective data management system. Yet the real impact of SCSS has not been explored. With these data, SCSS should track in its data base patterns that may be of safety concern to the nuclear power industry; that is, it should do what safety experts are doing manually. By following the patterns of concern, the system should be able to identify "precursor events," alert its users, and also alter its focus to important and relevant data entries in the data base.

Thus the real value of SCSS and its supporting technology is in the "information" that is imbedded not only in the data base but also in the interaction with its user. Of further real value is the relationship between the massive number of entries in the data base and the individual users. The same principle holds true with the way technology has spawned new relationships in financial service networks and news media.[1] Increasingly, the value resides in the communities of shared interest, not in the reams of data created by these technologies.

INTELLIGENT INFORMATION RETRIEVAL SYSTEMS

In an intelligent information retrieval system, information is the derivative of the relationship. This system, through a feedback technique that is based on neural networks, captures the topics in which its user is interested. On the basis of this information, the system learns the habits of its user by discovering the underlying patterns and collects and presents related news topics from its accessible resources. On the basis of this principle, the Universities of Texas and New Mexico have developed a knowledge robot (KNOWBOT) that works with nuclear data bases.[4]

Information in KNOWBOT is represented as units and their connections. These units participate in a cooperative environment in which information is encoded in the connection patterns among them. In general, the units may correspond to conceptual primitives, or they may have no particular meaning as individuals. The connection of the units reflect the association among them. Each connection is assigned a weight, which encodes the knowledge of how the units fit together in some domain.

This epistemic model is a directed acyclic graph (DAG) whose nodes represent the units of information (Fig. 1). Each unit $v \in V$ forms a node with a set of "parent" nodes Pa(v), where V is the set of all nodes in the network, and for each $w \in pa(v)$ a directed link $w \rightarrow v$ exists.[5] Conditional probability statements p(V|W) that are assigned to each link define the relationship of each node with its neighbors in the network.

Within this framework, then, the epistemic state of the system may be represented by

$$P = (D, T, C, T^{+})^{a6}$$

where $D = \{d_1, d_2, ..., d_n\}$ represents the set of all possible events of interest (e.g., shutdown, scram, etc.)

$T = \{t_1, t_2, ..., t_m\}$ represents the set of all possible manifestations that are reported to the data base (e.g., reports in the data base)

C = relation consisting of ordered pairs of causes and manifestations with $(d_i, t_j) \in C$

T^+ = the manifestations that have been instantiated for a given circumstance

One goal of KNOWBOT is to find the probability of each $d_i \in D$ given T^+. Once the set T^+ is instantiated by the user, the nodes in the network exchange a series of messages to update their states and those of the network. The framework for this update process is the Bayes' theorem:

$$p(d_i|T^+) = \frac{p(T^+|\, d_i)\, p(d_i)}{p(T^+|\, d_i)\, p(d_i) + p(T^+|\, \bar{d}_i)\, p(\bar{d}_i)}$$

The impact of introduction of each T^+ by the user may be viewed as a perturbation that propagates through the network via message-passing between neighboring variables.[7] Therefore KNOWBOT is an evolving environment that consists of the autonomous processors (the nodes in the DAG) and their interconnections. The design of the system is also based on the fact that a user is normally concerned with a small subset of a large data base; this is highly reflective of the user's domains of interest. The evolution of this environment is a continual process of

- The development of new or modification of existing units as new information becomes available
- Communication of the units in the environment
- The removal of inactive units that are no longer needed

Example

By extending this associative information retrieval system to relational data bases, the units correspond to

[a] A more general description of the system state may actually be given by $P = (D, T, C, T^+, T^-)$, where T^- represents the variables that are instantiated for their lack of existence. Therefore $T^+ \cup T^-$ represents the complete set of variables that are instantiated.

the attribute-value pairs that form the "tuples" in a given table. For example, two tuples of Table 1 are shown in Fig. 2. Further, each tuple in the table represents a "relationship" among a set of values.[8] Each value is a member of a domain that is defined for each column (attribute) of the table. Therefore, the node "Target-Rock" in Fig. 2 represents a value that appears in the manufacturer column of the table. The information about the associations that exist among the tuples and the attribute-value pairs is preserved in the connections among the units. The unit "Target-Rock," for example, is connected to both tuples 1 and 2 in Fig. 2.

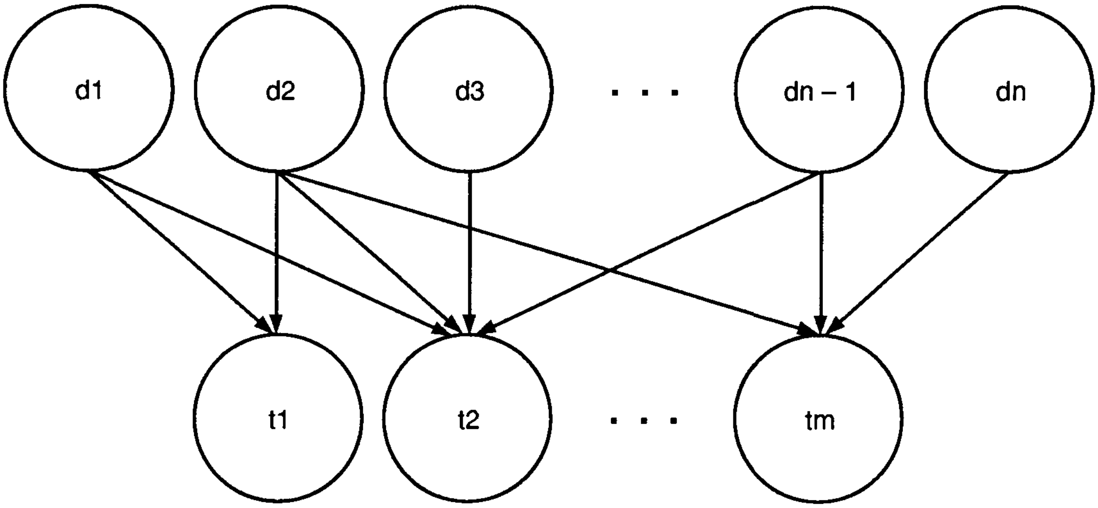

Fig. 1 The DAG that represents the structure of knowledge representation in KNOWBOT.

Table 1 Partial Listing of the NPRDS-PRV Data Base

Plant	Manufacturer	Model No.	Description
Nine Mile Point 1	Electromatic		Leak probable spring fatigue
Nine Mile Point 1	Electromatic		Leak lapped and rebuilt
Browns Ferry 3	Target-Rock	Three-Stage Model-67F	Leakage
Nine Mile Point 1	Electromatic		Valve flange gasket leak
Brunswick 1	Target-Rock	Three-Stage Model-67F	Pilot leak
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Bellows leaks
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Pilot leak
Peach Bottom 3	Target-Rock	Three-Stage Model-67F	Pilot leak
Pilgrim	Target-Rock	Three-Stage Model-67F	Pilot leak
Millstone 1	Target-Rock	Three-Stage Model-67F	Pilot leak
Hatch 1	Target-Rock	Three-Stage Model-67F	Pilot leak
Dresden 2	Electromatic		Failed part leaking seal rings
Browns Ferry 1	Target-Rock	Three-Stage Model-67F	Leaks wire drawn
Pilgrim	Target-Rock	Three-Stage Model-67F	Pilot valve leakage
Monticello	Target-Rock	Three-Stage Model-67F	Leaks foreign material
Brunswick 2	Target-Rock	Three-Stage Model-67F	Leak steam cutting
Pilgrim	Target-Rock	Three-Stage Model-67F	Leak oxidation cleaned lapped
Quad Cities 1	Electromatic		Main disk assembly excessive l
Quad Cities 2	Electromatic		Disk guide corrosion both possible . .
Monticello	Target-Rock	Three-Stage Model-67F	Bellows O-ring leak suspected
Millstone 1	Target-Rock	Three-Stage Model-67F	Pilot leak steam cutting
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Pilot valve disk leakage machined

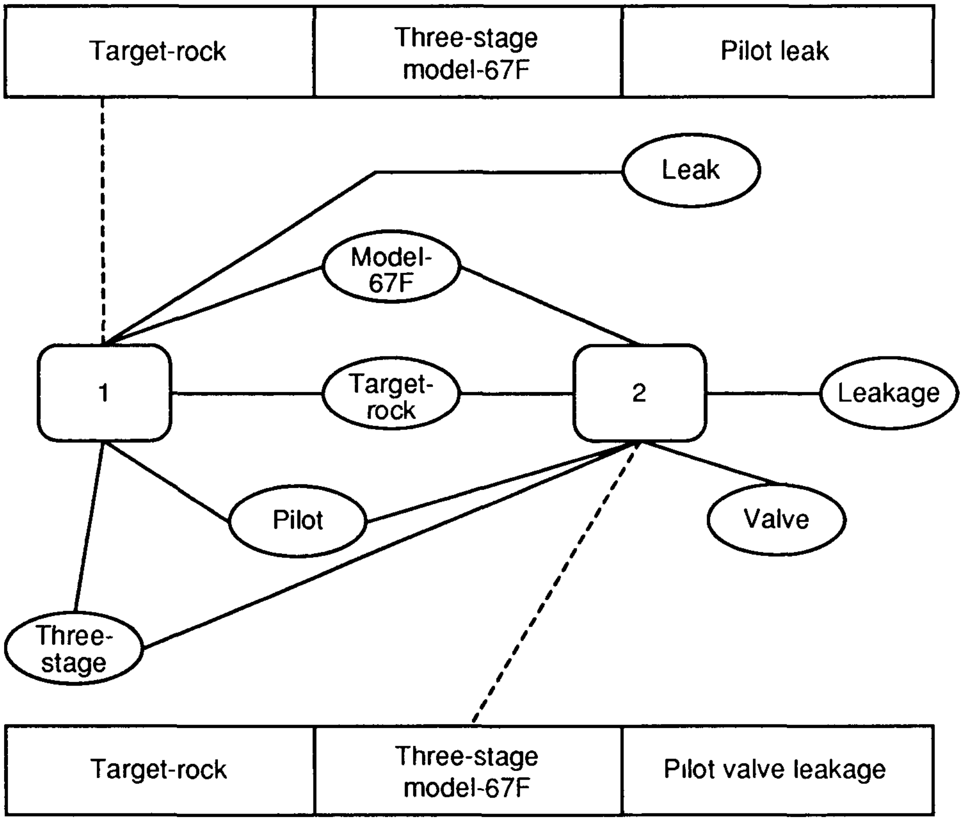

Fig. 2 Internal representation of tuples, their elements, and their interconnections in the associative information retrieval enviroment.

Test Data Base (NPRDS-PRV)

NPRDS-PRV was assembled to experiment with the interface and investigate its robustness. The data for the data base were extracted from a Department of Energy report on the analysis of the pressure-relieving valve failures using the NPRDS data. The data base consists of one relation, PRV-failure, with the following scheme: PRV-failure-scheme = (plant-name, manufacturer, model-number, description) The data base instance contains 82 tuples A partial display of the relation is shown in Table 1.

Case Study: Detection of Patterns of Concern

One of the intended usages of NPRDS, or other similar nuclear data bases, is to facilitate the identification of those individual events or generic situations which warrant additional analysis and evaluation One way an expert identifies a pattern or trend is by the *a priori* postulation of a concern. This concern could be entirely hypothetical. For example, the expert may wonder what has been the experience with the PRVs. On the other hand, he or she could postulate the concern on the basis of a nonspecific recall of information. For instance, he or she may remember observing many PRV failures at a given time. Under these conditions, the expert collects and reviews the relevant data to identify and isolate the events that relate to that concern After understanding the surrounding circumstances, he or she can evaluate the safety significance of the pattern [2]

With its induction feature, KNOWBOT can facilitate this objective as it will be demonstrated in the case study. Suppose a user is concerned about the failure events of the PRVs caused by leakage. One possible method is to access NPRDS-PRV for reports of PRV failures caused by leakage NPRDS-PRV has 22 tuples whose descriptions have direct references to the word leak or any combination of it (e.g., leak, leaks, leakage, etc.). There are, however, additional 27 tuples that imply the failure of the PRVs as the result of leakage. For example, the improper seating of the valve disks or their steam cutting implies that they had leaked. A conventional interface will retrieve the 22 records. The result of this search showing the PRV manufacturers, their model numbers, and the description of the failure events is shown in Table 1.

On the other hand, an expert may review these data and consequently access the data base with a refined query. For example, he or she may decide to search for those PRV failure reports that were caused by steam cut or improper seating. This modified query will produce additional reports that are indeed related to leak as the cause of failure. In a recursive process, the expert may retrieve all 49 reports from NPRDS-PRV. This extended study may, in turn, lead to the discovery of precursor events that may be cause for concern. With a conventional data base interface, the entire burden of this recursive search is on the human expert.

On the other hand, with KNOWBOT, the majority of this process can be carried out automatically. After the initial selection of the keys that directly relate to leak, the interface displays all 22 reports (see Table 1). But the interface, in an iterative process, continues to interrogate its environment for other related events. The screen will be updated to reflect the discovery of new records. At each update the records are sorted in the order of their correlation to the user request. The result of the search after several iterations by the interface is shown in Table 2. At any given instance, the user has the opportunity to redirect the search process by selecting new keys. This additional input represents his or her belief in the new discovery and acts as a positive reinforcement for KNOWBOT for its next iteration and search of the data base.

This inductive feature of KNOWBOT shares with the expert the burden of recursive searches in the data base. This case study shows that this feature can be a major contribution in discovering the patterns of concern in the data base and therefore an effective solution to the interface problem.

Table 2 Partial Listing of Result of an Inductive Search by KNOWBOT in the NPRDS-PRV Data Base for Leak-Related Failure Reports

Plant	Manufacturer	Model No.	Description
Brunswick 1	Target-Rock	Three-Stage Model-67F	Pilot leak
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Bellows leaks
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Pilot leak
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Pilot leak
Pilgrim	Target-Rock	Three-Stage Model-67F	Pilot leak
Millstone 1	Target-Rock	Three-Stage Model-67F	Pilot leak
Hatch 1	Target-Rock	Three-Stage Model-67F	Pilot leak
Dresden 2	Electromatic		Failed part leaking seal rings
Browns Ferry 1	Target-Rock	Three-Stage Model-67F	Leaks wire drawn
Pilgrim	Target-Rock	Three-Stage Model-67F	Pilot valve leakage
Monticello	Target-Rock	Three-Stage Model-67F	Leaks foreign material
Brunswick 2	Electromatic	Three-Stage Model-67F	Leak steam cutting
Pilgrim	Target-Rock	Three-Stage Model-67F	Leak oxidation cleaned lapped
Quad Cities 1	Electromatic		Main disk assembly excessive leakage
Quad Cities 2	Electromatic		Disk guide corrosion both possible
Monticello	Target-Rock	Three-Stage Model-67F	Bellows O-ring leak suspected
Millstone 1	Target-Rock	Three-Stage Model-67F	Pilot leak steam cutting
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Pilot valve disk leakage machined.
Nine Mile Point 1	Electromatic		Valve rings scored
Browns Ferry 3	Target-Rock	Three-Stage Model-67F	Did not reseat
Browns Ferry 1	Target-Rock	Three-Stage Model-67F	Wire drawn main seat
Monticello	Target-Rock	Three-Stage Model-67F	Pilot steam cutting
Millstone 1	Target-Rock	Three-Stage Model-67F	Pilot blow by
Hatch 1	Target-Rock	Three-Stage Model-67F	Failed bellows pressure switch
Pilgrim	Target-Rock	Three-Stage Model-67F	Steam cutting
Peach Bottom 2	Target-Rock	Three-Stage Model-67F	Galled steam binding in bushing

CONCLUSIONS

One of the most useful applications of NPRDS is to identify those individual events or generic situations which warrant additional analysis and evaluation. Therefore the expertise and natural pattern-recognition talents of human experts have been the two most important resources. The pattern-recognition capabilities of KNOWBOT can be utilized in cooperation with the human experts to detect these patterns.

The unpleasant reality is that most data base management systems are more interested in getting the needed data out in pleasant formats; they disregard the risk of user frustration and misunderstanding or of missing the true message that may be hidden in the data. These systems do not take advantage of the dynamic source of user–system interface to adapt their systems to the needs of the users.

Instead of asking, "What are the data that matter, and how are they most effectively managed?" these data base management systems must start asking, "What are the relationships that matter and how can the technology most effectively support them?"[1] This requires a totally different system design emphasis similar to that of KNOWBOT.

ACKNOWLEDGMENTS

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REFERENCES

1 M. Schrage, in Information Technology, the Key is Good Relationships, *The Wall Street Journal,* p. A18, Mar. 19, 1990.
2 F J. Hebdon, *Licensee Event Report System Description of System and Guidelines for Reporting,* Report NUREG-1022, 1983
3. N. M. Green, G. T Mays, and M P. Johnson, *Sequence Coding and Search System for Licensee Event Reports,* Report NUREG/CR-3905, 1985
4. A. S Heger and B. V. Koen, KNOWBOT: An Adaptive Database Interface, *Nucl Sci Eng* , 107(2) (1991).
5. D. J. Spiegelhalter and S. L. Lauritzen, Sequential Updating of Conditional Probabilities on Directed Graphical Structures, *Networks,* 20(5). 579-605 (August 1990).
6. R E. Neapolitan, *Probabilistic Reasoning in Expert Systems Theory and Algorithms,* John Wiley & Sons, Inc , New York, 1990
7 J Pearl, *Probabilistic Reasoning in Intelligent Systems Networks of Plausible Inference,* Morgan Kaufmann Publishers, San Mateo, Calif , 1988
8. H Korth and A. Silberschatz, *Database System Concepts,* McGraw-Hill, New York, 1986.

Technical Note: The Interagency Nuclear Safety Review Panel's Evaluation of the Ulysses Space Mission

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Abstract: *The October 1990 launch and deployment of the nuclear-powered Ulysses spacecraft from the Space Shuttle Discovery culminated an extensive safety review and evaluation effort by the Interagency Nuclear Safety Review Panel (INSRP) After more than a year of detailed independent review, study, and analysis, the INSRP prepared a Safety Evaluation Report (SER) on the Ulysses mission in accordance with Presidential Directive/National Security Council Memorandum 25 The SER, which included a review of the Ulysses Final Safety Analysis Report (FSAR) and an independent characterization of the mission risks, was used by the National Aeronautics and Space Administration (NASA) in its decision to request launch approval as well as by the Executive Office of the President in arriving at a launch decision based on risk–benefit considerations This paper provides an overview of the Ulysses mission and the conduct, as well as results, of the INSRP evaluation Although the mission risk determined by the INSRP in the SER was higher than that characterized by the Ulysses project in the FSAR, both reports indicated that the radiological risks were relatively small In the final analysis, the SER proved to be supportive of a positive launch decision The INSRP evaluation process has demonstrated its effectiveness numerous times since the 1960s In every case it has provided the essential ingredients and perspective to permit an informed launch decision at the highest level of our government*

An extensive flight safety review is required, per a Presidential Directive,[1] each time the United States plans to launch a spacecraft using a nuclear power source. The review, which culminates in an independent evaluation of the radiological risk of the mission by an Interagency Nuclear Safety Review Panel (INSRP), is documented in a Safety Evaluation Report (SER). The SER serves as a key element in the Presidential risk–benefit launch decision. The U.S. flight safety review and launch approval process for nuclear-powered space missions, described by Sholtis et al.,[2] was applied to the Ulysses mission from September 1989 to September 1990.

THE ULYSSES MISSION AND NUCLEAR POWER SYSTEM

The Ulysses mission is a joint endeavor of the European Space Agency and the National Aeronautics and Space Administration (NASA) to study the sun and its polar regions. The mission began with a daytime launch of the spacecraft aboard the Space Shuttle Discovery from the Kennedy Space Center, Florida, on Oct. 6, 1990. Shortly after being deployed from the Space Shuttle Orbiter, a two-stage Inertial Upper Stage (IUS) booster and a Payload Assist Module–Special Class booster propelled the spacecraft from an Earth parking orbit into an escape trajectory toward Jupiter. The transit time for the spacecraft to arrive at Jupiter is approximately one year and four months. Near Jupiter, the spacecraft will receive a gravity assist that will propel the spacecraft into a solar orbit that descends out of the ecliptic plane of the solar system. The trajectory will carry the spacecraft past the South Pole of the Sun during May–September 1994 and over the North Pole of the Sun one year later. Although the mission officially ends in September 1995, the spacecraft will remain in an elliptical orbit around the Sun with a perihelion of approximately 1.3 astronomical units (AU) and an aphelion of about 5.0 AU.

Because the Ulysses mission involves a Jupiter flyby, solar power was not practicable, and a nuclear power system was selected. Specifically, a single General Purpose Heat Source-Radioisotope Thermoelectric Generator (GPHS-RTG) containing approximately 11 kg of Pu-238 oxide provides the prime source of electric power for the Ulysses mission. The quantity of radioactive material contained in this GPHS-RTG necessitated an independent evaluation of the radiological risk of the Ulysses mission by the INSRP.

THE INSRP REVIEW

The scope of the INSRP review included consideration of accidents that could potentially result in the release of plutonium fuel into the environment during prelaunch operations, launch, ascent, on-orbit deployment, orbit insertion, and the Earth escape trajectory. To fulfill its responsibility, the INSRP and its five subpanels first reviewed the body of pertinent safety analysis reports and test data. The Ulysses Final Safety Analysis Report (FSAR) (Ref. 3) served as the prime input for the

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INSRP review. On the basis of this review, specific areas were identified for further study. The INSRP then conducted independent analyses. Those efforts resulted in the resolution of many issues, but some remained and were deemed to require alternative treatment. Those remaining issues were treated by the INSRP through the development and use of alternative assumptions, models, or interpretation of data. These alternative positions were then incorporated into the various computer codes and calculational routines as modifications. Finally, baseline and sensitivity calculations were conducted to determine the collective effect of the modifications made.

In all, the INSRP analyzed 19 accidents associated with the Ulysses mission, each of which had the potential for fuel release to the environment. One hundred thousand computer trials were performed for each accident to obtain fuel release amounts and probabilities. Of those 19 accidents, 11 were retained by the INSRP as "key" accidents for subsequent meteorological dispersion, health effects, and risk analysis. The eight accidents dropped from further consideration either had extremely small (≥2 mg) to no projected fuel releases to the environment or their overall probability of fuel release was extremely small ($\geq 10^{-9}$).

For the 11 key accidents carried through the complete analysis, two separate source terms were used—one representing an average fuel release amount and the other representing a fuel release amount characteristic of the tail of the fuel release distribution. This latter source term, labeled the "average of the top 5% source term," was obtained by averaging all the fuel release amounts above the 95th percentile from each of the accident fuel release distributions. A summary of the fuel release data obtained for the average source terms and the average of the top 5% source terms by accident type [e.g., random solid rocket booster (SRB), failure] and by mission elapsed time (MET), is provided in Tables 1 and 2, respectively.[4,5]

RESULTS

Summaries of the radiological health impacts obtained from the INSRP Ulysses evaluation are provided in Tables 3 (for the average source terms) and 4 (for the average of the top 5% source terms).

The INSRP also performed an integrated risk assessment in which treatment of both variance and uncertainty was incorporated, to determine and convey the state of knowledge about the radiological risks associated with the Ulysses mission more completely. The results of that assessment are illustrated in Figs. 1 to 3.

These results and the discussions that follow were taken from the Ulysses SER (Ref. 4).

DISCUSSION OF RESULTS

The overall mean calculated probability of an accident occurring through deployment and boost toward Jupiter, regardless of any considerations regarding fuel release, was on the order of 1 in 100. Given an initiating accident during the Ulysses mission, there was an additional (conditional) probability of failing one or more plutonia-fueled clads and releasing radioactive material into the environment. This would require either (1) an SRB failure that results in high-velocity fragments impacting the GPHS-RTG with sufficient energy to severely damage the fueled clads and release plutonia or (2) an explosion that results in hard surface ground impacts of GPHS-RTG hardware at or near terminal velocity. If an accident had resulted in reentry of the spacecraft during late ascent or from Earth orbit, the aeroshell modules were designed, and have been assessed, to withstand atmospheric reentry intact. For a fuel release to occur as a result of a reentry event, the aeroshell modules must subsequently strike hard surfaces. Such a release would be small and localized; thus it must occur in the immediate vicinity of people for exposures to occur.

No credible mechanism was identified that could result in a release of radioactive material prior to installation of the GPHS-RTG on the Ulysses spacecraft and the loading of propellants into the external tank of the Space Shuttle. In addition, once the spacecraft leaves the influence of the Earth's gravity toward Jupiter, no credible mechanism was identified that can return the spacecraft and its radioactive materials to the vicinity of Earth.

The most likely and thus the expected result for all accident scenarios was no fuel release and the expected outcome for the Ulysses mission was a successful launch and deployment.

An interesting finding of the INSRP evaluation was that a Challenger-type accident was projected to yield no fuel release to the environment.[5]

For each key accident scenario, two single-point source-term estimates were calculated: (1) an average source term and (2) an average of the top 5% source term. For the average source terms, the calculated number of cancer fatalities ranged from 0.002 with a probability of approximately 1 in 29 000 to 3 at approximately 1 in 1 000 000. For the average of the top 5% source terms, the calculated number of cancer fatalities ranged from 0.008 with a probability of about 1 in a million to 36 with a probability of less than 1 in 100 million. In all

Table 1 Summary of Fuel Releases for Key Accident Scenarios (Average Source Terms)[a]

Phase	MET	Accident type	Probability of initiating event	Conditional probability of fuel release	Aggregate fuel release probability	Source terms: Grams	Source terms: Curies	Release phenomena
0	T-6 h to T = 0	On-pad external tank explosion	2.9×10^{-3}	3.0×10^{-3}	8.6×10^{-6}	0.08 Ground	1.0	Coagulation and plume transport aloft
1	0 to 2 s	Tipover/tower impact	1.9×10^{-4}	4.4×10^{-3}	8.3×10^{-7}	0.06 Ground	0.8	Coagulation and plume transport aloft
1	0 to 10 s	Near pad external tank explosion	1.2×10^{-3}	2.4×10^{-3}	2.9×10^{-6}	0.1 Ground	1.2	Coagulation and plume transport aloft
1	0 to 10 s	Near-pad SRB random failure (air–ground release)	1.5×10^{-3}	3.6×10^{-3}	5.2×10^{-6}	2.0 Air 4.3 Ground	24 50	Air vaporization, coagulation, and plume transport aloft Ground coagulation, 4-m puff 2 m off the ground
1	0 to 10 s	Near-pad SRB random failure (ground release only)	1.5×10^{-3}	3.0×10^{-2}	4.5×10^{-5}	1.9 Ground	23	Coagulation, 4-m puff 2 m off the ground
1	10 to 20 s	Early ascent SRB random failure (air–ground release)	3.7×10^{-4}	3.8×10^{-3}	1.4×10^{-6}	1.4 Air 2.5 Ground	16 30	Air vaporization, coagulation, and plume transport aloft Ground 4-m puff 2 m off the ground
1	10 to 20 s	Early ascent SRB random failure (ground release only)	3.7×10^{-4}	4.6×10^{-3}	1.7×10^{-6}	8.5 Ground	100	Coagulation, 4-m puff 2 m off the ground
1	20 to 57 s	Early, mid-ascent SRB random failure	5.7×10^{-4}	2.8×10^{-3}	1.6×10^{-6}	1.2 Air 0.05 Ground	14 0.6	Air plume transport aloft Ground 4-m puff 2 m off the ground
1	57 to 105 s	Late, mid-ascent SRB random failure	3.6×10^{-4}	4.2×10^{-3}	1.5×10^{-6}	6.0 Air	72	Worldwide transport aloft
1	105 to 120 s	Late ascent SRB random failure	1.7×10^{-4}	2.1×10^{-2}	3.6×10^{-6}	23.7 Air	280	Worldwide transport aloft
2, 3, or 4	120 s til IUS burns complete	Inadvertent re-entry and land impact	1.7×10^{-3}[b]	3.6×10^{-1}	6.2×10^{-4}	0.032 Ground (rock)	0.4	4-m puff 2 m off the ground

[a]MET, mission elapsed time, IUS, inertial upper stage, SRB, solid rocket booster

[b]Includes the probability of an inadvertent reentry and, given reentry, that the General Purpose Heat Source modules hit land

cases, calculated fatalities were those which might be expected within the fifty-year period following an accident in which it is assumed that no intervention or mitigation is taken. (For health effects greater than one, the calculated fatalities were entirely due to high-altitude fuel releases that would result in extremely small doses to the world population. For such doses, the collective and individual risk increments are calculable but not demonstrable. In fact, the possibility of zero risk cannot be ruled out of a strict statistical analysis of data, especially when predicted risks are less than 10^{-5}. Consequently, an important point regarding these radiological risk increments or additions is frequently omitted (that is, these risks are expressions of a probability distribution and are not a certainty).

On the basis of the INSRP integrated risk assessment for the entire Ulysses mission, one can conclude with 95% confidence that the probability of one or more cancer fatalities was about 1 in 100 000, and the probability of 12 or more cancer fatalities was about 1 in a million. Similarly, one can conclude with 95% confidence that the likelihood of one or more cancer fatalities in local Florida was less than 1 in a million and that the likelihood of one or more cancer fatalities worldwide was about 1 in

Table 2 Summary of Fuel Releases for Key Accident Scenarios (Average of the Top 5% Source Terms)[a]

Phase	MET	Accident type	Probability of initiating event	Conditional probability of fuel release	Aggregate fuel release probability	Source terms: Grams	Source terms: Curies	Release phenomena
0	T-6 h to T = 0	On-pad external tank explosion	2.9×10^{-3}	1.5×10^{-4}	4.4×10^{-7}	0.23 Ground	2.7	Coagulation and plume transport aloft
1	0 to 2 s	Tipover/tower impact	1.9×10^{-4}	2.2×10^{-4}	4.2×10^{-8}	0.24 Ground	2.8	Coagulation and plume transport aloft
1	0 to 10 s	Near-pad external tank explosion	1.2×10^{-3}	1.3×10^{-4}	1.5×10^{-7}	0.32 Ground	3.8	Coagulation and plume transport aloft
1	0 to 10 s	Near-pad SRB random failure (air–ground release)	1.5×10^{-3}	1.7×10^{-4}	2.6×10^{-7}	32.1 Air 28.1 Ground	380 330	Air vaporization, coagulation, and plume transport aloft Ground coagulation, 4-m puff 2 m off the ground
1	0 to 10 s	Near-pad SRB random failure (ground release only)	1.5×10^{-3}	1.5×10^{-3}	2.3×10^{-6}	21.3 Ground	250	4 m puff 2 m off the ground
1	10 to 20 s	Early ascent SRB random failure (air–ground release)	3.7×10^{-4}	1.9×10^{-4}	6.9×10^{-8}	20.3 Air 11.8 Ground	240 140	Air vaporization, coagulation, and plume transport aloft Ground 4-m puff 2 m off the ground
1	10 to 20 s	Early ascent SRB random failure (ground release only)	3.7×10^{-4}	2.3×10^{-4}	8.4×10^{-8}	32.8 Ground	390	4-m puff 2 m off the ground
1	20 to 57 s	Early, mid-ascent SRB random failure	5.7×10^{-4}	1.4×10^{-4}	7.9×10^{-8}	20.9 Air 0.7 Ground	250 8.7	Air plume transport aloft Ground 4-m puff 2 m off the ground
1	57 to 105 s	Late, mid-ascent SRB random failure	3.6×10^{-4}	2.1×10^{-4}	7.5×10^{-8}	106 Air	1260	Worldwide transport aloft
1	105 to 120 s	Late ascent SRB random failure	1.7×10^{-4}	1.1×10^{-3}	1.8×10^{-7}	269 Air	3200	Worldwide transport aloft
2, 3, or 4	120 s til IUS burns complete	Inadvertent reentry and land impact	1.7×10^{-3}[b]	1.4×10^{-1}	2.3×10^{-4}	0.063 Ground (rock)	0.8[c]	Two 4-m puffs 2 m off the ground

[a]MET, mission elapsed time, IUS, inertial upper stage, SRB, solid rocket booster

[b]Includes the probability of an inadvertent reentry and, given reentry, that the General Purpose Heat Source modules hit land

[c]This involves two separate releases at two different locations each of 0.032 g or 0.375 Ci

100 000. (The breakpoint for effects in local Florida and worldwide effects occurs for projected fuel releases at a MET of approximately fifty seven seconds, when the launch vehicle reaches the stratosphere.)

To place the health-related risks calculated in the INSRP analysis in some perspective, a comparison with a similar type of exposure and risk is useful. Two such comparisons were provided. First, a comparison was made between the highest fifty-year dose calculated to be received by any individual and the radon background dose received by that same individual for the same time period. Second, a comparison was made between the natural occurrence of fatal cancer in the population and the highest added incremental cancer risk to any single individual.

It is generally accepted that, of the approximately 350 mrem average annual background radiation dose experienced by the population, approximately 0.2 rem (with a probability of 1) is due to naturally occurring radon daughter product exposure. Thus the lifetime (fifty-year) accumulated radon dose to an individual in the population would be 10 rem. If one compares this with the calculated fifty-year maximum dose of 0.21 rem (with a probability of less than 1 in 4 million) to the maximally exposed individual in the local Florida population, that individual would receive approximately 2% of

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Table 3 Radiological Health Impact (Average Case)

Phase	Accident type	Air–ground source term, Ci	Release probability	Maximum individual dose, rem (over 50 yr)	Population potentially exposed	Collective dose, person-rem		Collective organ dose, person-rem				Total health effects	Frequency of health effects in events/mission
						1st yr	50 yr	Lung	Liver	Bone	RBM[a]		
0	On-pad external tank explosion	–/1 0	$8\,6 \times 10^{-6}$	$7\,8 \times 10^{-4}$	$6\,3 \times 10^{5}$	18	28	0 013	59	330	27	0 007	$3\,5 \times 10^{-6}$
1	0 to 2 s tipover/tower impact	–/0 8	$8\,3 \times 10^{-7}$	$5\,7 \times 10^{-4}$	$6\,3 \times 10^{5}$	17	26	0 01	58	340	27	0 007	$3\,2 \times 10^{7}$
1	0 to 10 s near-pad external tank explosion	–/1 2	$2\,9 \times 10^{-6}$	$9\,4 \times 10^{-4}$	$6\,3 \times 10^{5}$	20	32	0 015	65	370	29	0 008	$1\,2 \times 10^{-6}$
1	0 to 10 s near-pad SRB random failure (air–ground release)	24/50	$5\,2 \times 10^{-6}$	$2\,0 \times 10^{-2}$	$7\,4 \times 10^{5}$	460	610	320	1400	8100	650	0 2	$2\,7 \times 10^{-6}$
1	0 to 10 s near-pad SRB random failure (ground release only)	–/23	$4\,5 \times 10^{5}$	$7\,0 \times 10^{-3}$	$6\,1 \times 10^{5}$	160	188	160	530	3100	250	0 07	$6\,3 \times 10^{-6}$
1	10 to 20 s early ascent SRB random failure (air–ground release)	16/30	$1\,4 \times 10^{-6}$	$9\,7 \times 10^{-3}$	$1\,3 \times 10^{6}$	240	320	350	610	3400	270	0 09	$7\,0 \times 10^{-7}$
1	10 to 20 s early ascent SRB random failure (ground release only)	–/100	$1\,7 \times 10^{-6}$	$2\,2 \times 10^{-2}$	$6\,2 \times 10^{5}$	500	590	490	1600	9600	770	0 2	$7\,0 \times 10^{-7}$
1	20 to 57 s early, mid ascent SRB random failure	14/0 6	$1\,6 \times 10^{-6}$	$2\,3 \times 10^{-4}$	$6\,2 \times 10^{5}$	6 7	130	18	17	93	75	0 05	$8\,8 \times 10^{7}$
1	57 to 105 s late, mid ascent[b] SRB random failure	72/	$1\,5 \times 10^{-6}$		Worldwide		2900					0 08	$1\,1 \times 10^{-6}$
1	105 to 120 s late ascent[b] SRB random failure	280/	$3\,6 \times 10^{-6}$		Worldwide		$1\,1 \times 10^{4}$					3 00	$9\,1 \times 10^{-7}$
2, 3, or 4	Inadvertent reentry and land impact	–/0 4	$6\,2 \times 10^{-4}$	$8\,6 \times 10^{1}$	$1\,96 \times 10^{3}$	5 9	7 2					0 002	$3\,5 \times 10^{5}$

[a]RBM, red bone marrow

[b]Assumes 40 person-rem/Ci, based on Ref 6, and $2\,9 \times 10^{-4}$ health effects per person-rem for air releases

Table 4 Radiological Health Impact (Average of Top 5% Source Terms)

Phase	Accident type	Air–ground source term, Ci	Release probability	Maximum individual dose, rem (over 50 yr)	Population potentially exposed	Collective dose, person-rem		Collective organ dose, person-rem				Total health effects	Frequency of health effects in events/mission
						1st yr	50 yr	Lung	Liver	Bone	RBM [a]		
0	On-pad external tank explosion	–/2 7	4 4 × 10^{7}	2 0 × 10^{3}	6 3 × 10^{5}	42	66	0 033	130	760	60	0 02	~1 × 10^{-7}
1	0 to 2 s tipover/tower impact	–/2 8	4 2 × 10^{-8}	2 2 × 10^{-3}	6 3 × 10^{5}	49	76	0 036	160	900	71	0 02	<1 × 10^{-8}
1	0 to 10 s near-pad external tank explosion	–/3 8	1 5 × 10^{-7}	3 0 × 10^{-3}	6 3 × 10^{5}	63	98	0 048	200	1100	89	0 02	~3 × 10^{8}
1	0 to 10 s near pad solid rocket booster (SRB) random failure (air–ground release)	380/330	2 6 × 10^{7}	2 1 × 10^{1}	7 0 × 10^{5}	3000	4100	2900	7500	4 4 × 10^{4}	3200	0 9	<1 × 10^{-8}
1	0 to 10 s near-pad SRB random failure (ground release only)	–/250	2 3 × 10^{-5}	8 9 × 10^{-3}	6 1 × 10^{5}	1500	1700	2200	4500	2 6 × 10^{4}	2100	0 6	<1 × 10^{-8}
1	10 to 20 s early ascent SRB random failure (air–ground release)	240/140	6 9 × 10^{-8}	4 4 × 10^{-2}	1 2 × 10^{6}	1200	1600	1300	3000	1 6 × 10^{4}	1300	0 4	<1 × 10^{8}
1	10 to 20 s early ascent SRB random failure (ground release only)	–/390	8 4 × 10^{8}	1 1 × 10^{1}	6 1 × 10^{5}	2300	2600	3200	7000	4 1 × 10^{4}	3300	0 9	<1 × 10^{8}
1	20 to 57 s early, mid-ascent SRB random failure	250/8 7	7 9 × 10^{8}	2 1 × 10^{3}	6 4 × 10^{5}	64	1800	160	160	880	71	0 6	<1 × 10^{-8}
1	57 to 105 s late, mid ascent[b] SRB random failure	1260/–	7 5 × 10^{-8}		Worldwide		5 0 × 10^{4}					14	<1 × 10^{8}
1	105 to 120 s late ascent[b] SRB random failure	3200/–	1 8 × 10^{7}		Worldwide		1 3 × 10^{5}					36	<1 × 10^{8}
2, 3, or 4	Inadvertent reentry and land impact	–/0 8	2 3 × 10^{-4}	3 3	1 0 × 10^{4}	23	28					0 008	~1 × 10^{-6}

[a] RBM, red bone marrow

[b] Assumes 40 person-rem/Ci, based on Ref 6, and 2 9 × 10^{-4} health effects per person-rem for air releases

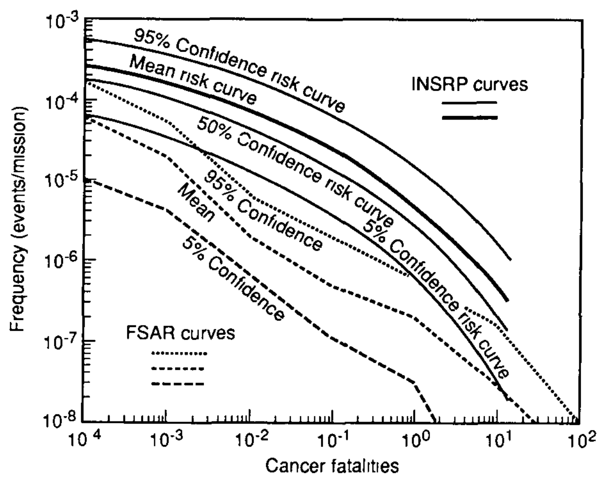

Fig. 1 Approximation of total mission radiological risk by sum of all scenarios. INSRP, Interagency Nuclear Safety Review Panel; FSAR, Final Safety Analysis Report.

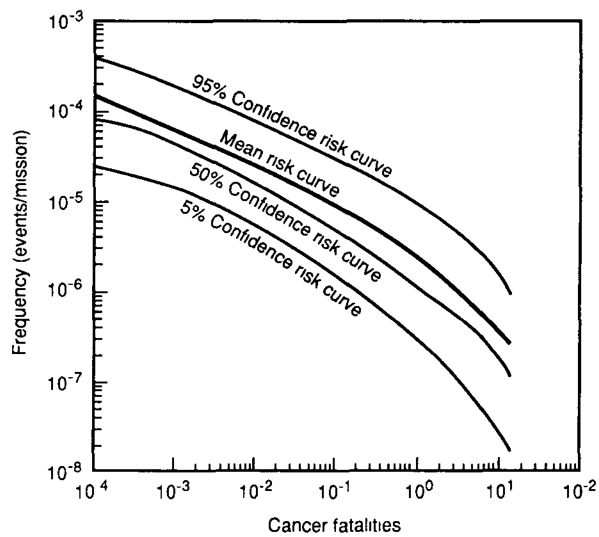

Fig. 3 Approximation of worldwide radiological risk by sum of all scenarios after 57-s mission elapsed time.

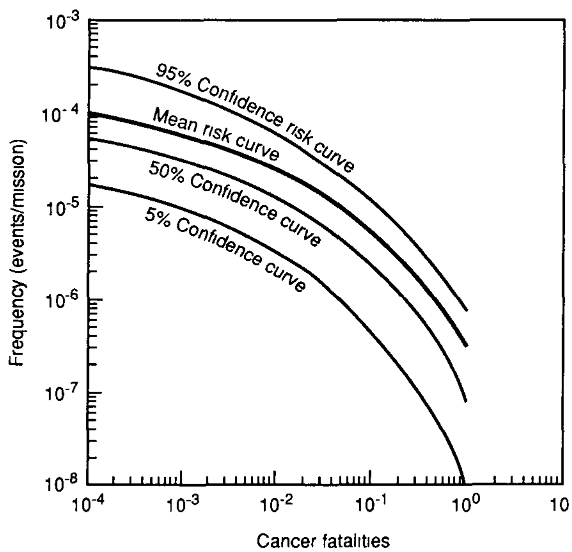

Fig. 2 Approximation of local Florida radiological risk by sum of all scenarios prior to 57-s mission elapsed time.

the radon background. In the case of the maximum fifty-year individual dose of 3.3 rem (with a probability of much less than 1 in 4 000) calculated for the maximally exposed individual in the world population, that individual would receive approximately 33% of the radon background. Calculated exposures to the remaining population would be a small fraction of these percentages.

Compared with the nominal 20% lifetime cancer fatality risk that everyone faces, the highest calculated added individual risk associated with the Ulysses mission increased lifetime cancer risk to no more than 20.00015%. If one considers that the mean likelihood of an accidental release that results in fatal cancer was less than 1 in 100 000, the actual added risk of fatal cancer associated with the Ulysses mission was much smaller than 0.00015%. Thus the INSRP analysis suggested that the radiological risks associated with the Ulysses mission were relatively small.

Although the mission risk determined by the INSRP in the SER was higher than that characterized by the Ulysses project in the FSAR, as illustrated in Fig. 1, both reports indicated that the radiological risks were relatively small. In the final analysis, the SER proved to be supportive of a positive launch decision.

The INSRP evaluation process has demonstrated its effectiveness more than 20 times since the 1960s. In every case it has provided the essential ingredients and perspective to permit an informed launch decision at the highest level of our government.

ACKNOWLEDGMENTS

We thank the INSRP subpanel members, current and past, as well as previous INSRP coordinators. These individuals spent countless hours completing the INSRP subpanel reports and the Ulysses SER. Without the diligence and assistance of these dedicated individuals, the INSRP review of the Ulysses space mission could not have been completed and the Ulysses SER, as well as this report, would not have been possible.

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REFERENCES

1 The White House, *Scientific or Technological Experiments with Possible Large Scale Adverse Environmental Effects and Launch of Nuclear Systems into Space,* Presidential Directive/National Security Council Memorandum 25, PD/NSC-25, December 14, 1977

2 J A Sholtis, Jr, J P Joyce, and R C Nelson, U S Flight Safety Review/Approval Process for Nuclear-Powered Space Missions, in *Proceedings of the Seventh Symposium on Space Nuclear Power Systems*, Report CONF-900109, pp 569-571, 1990

3 Department of Energy, *Final Safety Analysis Report for the Ulysses Mission,* Summary, ULS-FSAR-001, Vol 1, ULS-FSAR-002, Vol 2, Book 1, ULS-FSAR-003, Vol 2, Book 2, ULS-FSAR-004, Vol 3, Book 1, ULS-FSAR-005, Vol 3, Book 2, ULS-FSAR-006, March 1990

4 Interagency Nuclear Safety Review Panel, *Safety Evaluation Report for Ulysses* Report INSRP 90-01, Vols I, II, and III, July 1990 (predecisional), September 1990

5 Interagency Nuclear Safety Review Panel, *Power System Subpanel Report for Ulysses* Report INSRP 90-04, July 1990 (predecisional), September 1990

6 International Commission on Radiological Protection, Publication 48, *The Metabolism of Plutonium and Related Elements,* 1986

Accident Analysis

Edited by R. P. Taleyarkhan

The Severe Accident Analysis Program for the Savannah River Nuclear Production Reactors

By M. L. Hyder[a]

Abstract: *Severe accident phenomena pertinent to the heavy-water-moderated production reactors of the U.S. Department of Energy are being studied in the Severe Accident Analysis Program (SAAP) at the Savannah River Site. The SAAP has sought to define the behavior of the Savannah River reactors in accident scenarios involving significant fuel melting. The goal of the program is to make possible accident analyses of the production reactors that are of comparable quality to those done for power reactors.*

These large Savannah River reactors differ from power reactors in several important respects: they operate at low temperature and pressure, their fuel is uranium metal alloyed and clad with aluminum, and radioactive releases are contained by a filtered confinement system rather than by static containment. These differences have guided the experimental and calculational development of the SAAP, which also draws where possible on analyses made for other types of reactors.

Major experimental research areas in the SAAP have included fuel-melting phenomena, melt relocation and its interactions with water and concrete, fission-product release and mobility, and the response of the confinement system under accident conditions. The MELCOR and SCDAP/RELAP5 code packages, developed for severe accident analyses of commercial power reactors, have also been adapted to the Savannah River reactors under the SAAP. These are already being used in accident analyses. Calculational tools have also been developed for estimating the potential effects of steam explosions. Experimental work is continuing, as is the validation and application of the code packages.

At the Savannah River Site (SRS) near Aiken, S.C., nuclear materials, including plutonium and tritium, are produced for the U.S. Government in nuclear reactors constructed for this purpose. These reactors are of unique design; they use tubular elements of highly enriched uranium–aluminum fuel moderated and cooled by heavy water. All were built in the early 1950s. Before a recent extended shutdown, three were still being operated. The site and reactors are operated under contract to the U.S. Department of Energy (DOE) by the Westinghouse Savannah River Company.

The Savannah River reactors differ markedly from commercial power reactors. Although the SRS reactors have been operated at a thermal power comparable to commercial power reactors, the coolant is maintained below the boiling point at near atmospheric pressures. This permits the use of aluminum-based fuel, target, and core structure components. The SRS reactors, which were constructed before the development of containment structures for reactors, incorporate instead a once-through ventilation system with filtration and iodine absorption treatment of the off gas.

A schematic diagram of the coolant system of SRS reactors is shown in Fig. 1. Heavy water, which serves as both moderator and coolant, enters the coolant plenum at the top of the vessel through six coolant loops, one of which is illustrated. From the plenum, the heavy water flows downward through more than 400 tubular fuel assemblies. A cross section of a typical assembly, which includes two fuel and two target tubes, is shown in Fig. 2. At the bottom of each tube the heated effluent flows out into the moderator surrounding the assemblies before being recycled through heat exchangers. Each assembly is continuously monitored for flow and effluent temperature by sensors in the monitor pin.

[a]Westinghouse Savannah River Company, Savannah River Laboratory.

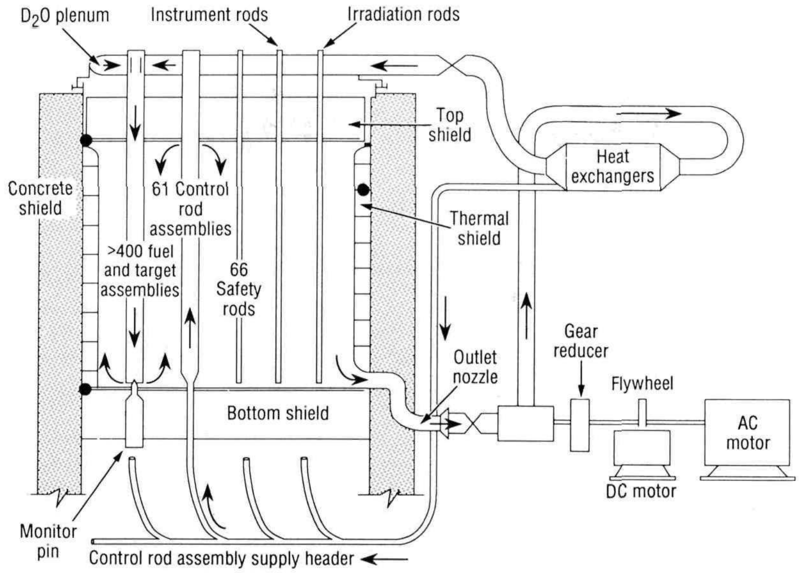

Fig. 1 Schematic diagram of the reactor coolant system.

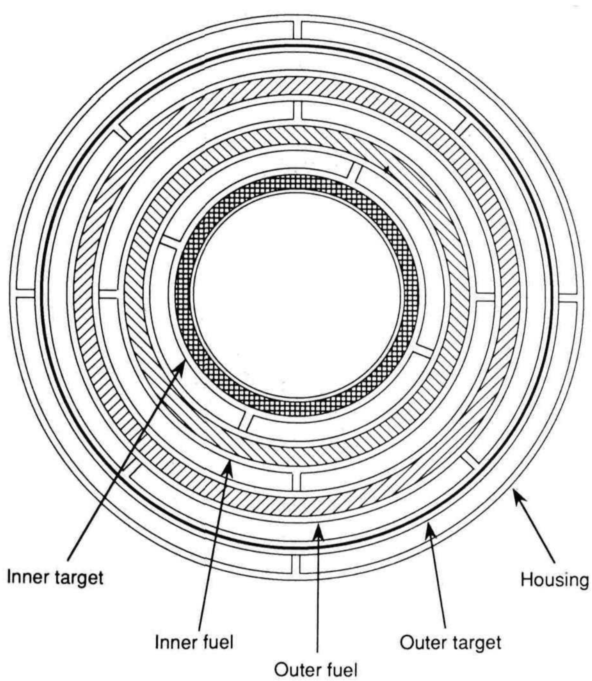

Fig. 2 Typical Savannah River fuel assembly (cross section).

In a severe accident in which fuel is overheated, radioactive isotopes would be released into the primary coolant, and volatile isotopes could gradually reach the building air through pressure-relief devices. If the accident involved both loss of coolant and fuel damage, contaminated coolant would reach the floor and sumps in the bottom of the building and thus release further radioactivity to the building. Protection against release of most radioactivity is provided by the Airborne Activity Confinement System. In this system all effluent air passes through up to five parallel filter compartments before release through the stack. The interior of the filter compartments is shown in Fig. 3. Each of the three filtration components shown occupies an area 2.44 m (8 ft) wide and 4.87 m (16 ft) high. Moist and some solid aerosols are removed by moisture separator prefilters; remaining aerosols are removed by high-efficiency particulate air (HEPA) filters. Gaseous iodine is removed by sorption on activated carbon. The remaining activity, which is nearly all noble gases, is released and dispersed by a 61-m (200-ft) stack.

Since the construction of the SRS reactors in the early 1950s, several internal reviews were made of their safety, and various safety features, including the Airborne Activity Confinement System, have been retrofitted. This activity paralleled the increasing attention to safety and accident studies in the burgeoning commercial nuclear industry. Following the accident at Three Mile Island Nuclear Station, extensive studies of reactor safety were made by the Nuclear Regulatory Commission and other organizations. These were concentrated on commercial

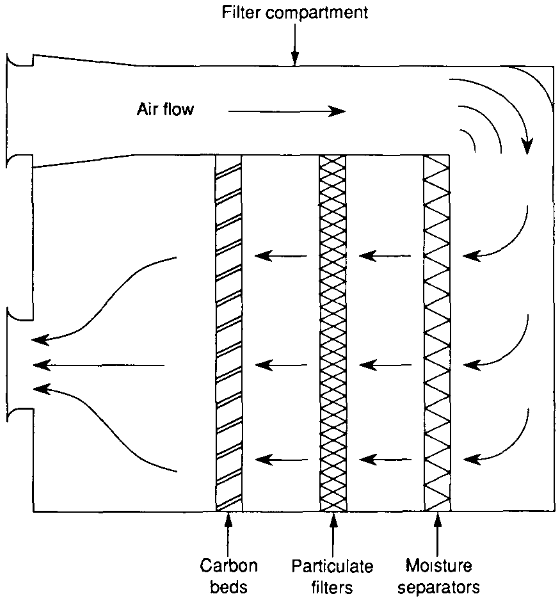

Fig. 3 Filter compartment (elevation cross section).

power reactors, which are cooled and moderated by light water and are fueled with zirconium-clad uranium oxide fuels of moderate enrichment. The results showed the importance of design-specific safety features to the potential risk of accidental releases of radioactivity.

Recognition of the substantial design differences between power reactors and the SRS production reactors led the site contractor to undertake a substantial program of safety studies for the latter. This was conducted by the Savannah River Laboratory (SRL), the research and development branch of the Savannah River Site organization. Safety activities included extensive reviews and testing of reactor components and systems, the development of a full-scope Probabilistic Risk Assessment (PRA),[1] and a calculational and experimental study of severe accident phenomena. The last activity, the Severe Accident Assessment Program (SAAP), is the subject of this article. In general, accidents producing these phenomena are beyond the design basis for the Safety Analysis Report for these reactors, but an understanding of them is important to the overall assessment of risk. They include, for example, accidents involving loss of primary or secondary coolant or loss of the pump capacity needed for cooling. The PRA has identified the accident sequences of greatest concern and has also helped identify the physical phenomena needing experimental investigation.

The SAAP began in 1987, following an internal study of programmatic needs. Its purpose is to provide an understanding of accident phenomena in SRS reactors comparable with that obtained with light-water power reactors. Its intention is specifically to develop accident information peculiar to SRS reactors, the needs for which were identified by internal reviews and outside consultants. Experimental subjects addressed in the program include the physical behavior of molten fuel and its interactions with water and with structures; the release rates of fission products and their chemical forms; the migration of fission products through the reactor building and filtration system; and energetic events, such as steam explosions. This information is used in safety documentation for the SRS reactors, including the PRA. It has also been important in supporting restart of the reactors following the recent shutdown and safety review. The program includes two major activities: the development of severe accident computer codes for SRS reactors and the characterization of severe accident phenomena peculiar to those reactors.

The codes are most immediately usable for analysis and evaluation in the PRA and other safety analysis, but the experimental work is essential to understand the phenomena involved and to provide validation. The effort devoted to the two activities has been roughly comparable. Both experimental and code development activities have largely been done by other laboratories under subcontract to SRS because of the limited resources available to the program at Savannah River and the recognition that in many areas the most effective way to proceed is to use existing expertise. Program guidance was obtained in the beginning from other laboratories experienced in severe accident studies and subsequently from a review committee of outside experts that has met periodically to review the course of the program. Whereas the program was originally envisioned as a four-year effort extending through 1991, all aspects were not specified at that time, and it has continued to develop. Additionally, a continuing effort beyond that time will be necessary to keep up with advancing knowledge in this field.

The first extensive publication of the results from the SAAP was included in proceedings of the meeting of the American Nuclear Society on Non-Power Producing Reactors in Boise, Idaho, during October 1990 (Ref. 2). A substantial number of technical reports, journal articles, and other papers are currently in preparation, and some have been issued; these will be cited where possible.

In 1989, an additional program of severe accident studies was initiated by the DOE in support of the proposed heavy-water new production reactor proposed for construction during the 1990s. This program, which is administered through Sandia National Laboratories (SNL), is complementary to the SAAP in several areas. These include particularly the development of techniques for calculating accident progression with possible nuclear criticality and confirmatory in-pile melt testing. Contacts between the programs have been established for coordination and the exchange of results.

In the following discussion, the code development and experimental studies of the SAAP are considered in turn.

COMPUTER CODE DEVELOPMENT

The principal computer code development activities in the SAAP have been the modification of MELCOR (Ref. 3) and of SCDAP/RELAP (Ref. 4) for use with the SRS reactors. The former can be used to simulate complete accident sequences, including core behavior, activity release and movement, and release from the building. The latter provides a more detailed treatment of accident conditions within the primary coolant system. Additionally, a code named K-FIX(GT) (Ref. 5) was developed to determine the effects of steam explosions resulting from contact between fuel melts and water.

MELCOR

The MELCOR severe accident code, originally developed by SNL for light-water reactors (LWRs), was modified for use with SRS reactors by Science Applications International Corporation (SAIC). The new code, referred to as MELCOR/SR-Mod 3, was written to be consistent with MELCOR 1.8.0, the current version during 1990 (Ref. 3). Of the new models developed for this code, the core and ventilation models are the most important since these components of the reactor differ considerably from commercial reactors and are very important in severe accidents. A description of these models was given at the Boise meeting.[6,7]

Although MELCOR/SR-Mod 3 was not completed until late in 1990, portions were available earlier and were used in the SRS Reactor PRA. Parametric studies with the completed code, and the core model in particular, are now in progress. Note that one of the modifications includes entering the properties of heavy water (D_2O). Although the differences between the properties of H_2O and D_2O are not large, they need to be considered explicitly in calculations of hydrogen burns and when viscous forces predominate.

An updated version of MELCOR/SR-Mod 4, is to be issued in early 1992. It is being written to be consistent with the latest LWR version, MELCOR 1.8.1. MELCOR/SR-Mod 4 will be verified and validated by WSRC during 1992–1993.

MELCOR/SR-Mod 4 does not include a detailed fission-product and aerosol release model for SRS reactor core–concrete reactions. This will be addressed through modifications to an existing code, VANESA (Ref. 8), or by development of a new code.

SCDAP/RELAP

The SCDAP/RELAP series of severe accident and thermal–hydraulic codes originating at Idaho National Engineering Laboratory has been modified by its authors to describe severe accidents in the SRS reactor vessel. This is a mechanistic code set that involves a detailed nodalization of the reactor vessel and primary coolant system, including fuel and target components.[9]

Steam Explosion Code

Under the direction of Prof. S. Abdel-Khalik of the Georgia Institute of Technology, a computer program has been written to simulate the propagation of steam explosions.[5] (Steam explosions are not considered explicitly in the conventional severe accident codes, such as MELCOR, because of their complexity.) This code requires as input the initial parameters of the steam explosion (i.e., the energy and mass of the materials involved). It then models the expansion phase of the explosion in detail, including interactions with objects in the surroundings. It is based on a three-dimensional modification of the K-FIX (Ref. 10) code. The new code is intended for such tasks as predicting the effects of a large steam explosion outside the reactor vessel on the reactor building and fixtures. In conjunction with a suitable material response code, the response of walls and reactor components can be predicted.

EXPERIMENTAL STUDIES

The extensive experimental studies conducted under this program are grouped for convenience into categories: Accident progression, release and mobilization of radioactivity, energetic phenomena, confinement system operation and response, and materials and supporting studies. Each of these will be considered in turn.

Fuel Melting and Accident Progression

Melting of aluminum-based fuels has been studied numerous times over the years, but a consistent model of fuel-melt progression is still needed, and development of such a model is a major part of this program. In laboratory studies aluminum is frequently substituted for the aluminum alloys used in fuel and target materials. This is especially true in cases in which uranium might be dispersed by the experiment. The use of pure aluminum in many cases is not a bad approximation because the aluminum is generally present in large excess on the atomic scale. However, alloying can have a considerable effect on such phenomena as melting. Further complications arise from the presence of fission products in irradiated fuel. These can have both chemical and physical effects (such as inducing foaming) on the behavior of fuel under accident conditions.

As previously noted, typical SRS fuel assemblies consist of concentric tubes about 4.6 m (15 ft) in length. During operation, heavy-water coolant flows through the annuli between the concentric cylinders of fuel and target material. The fuel tubes consist of an alloy of aluminum and enriched uranium clad with aluminum on all surfaces; target tubes of lithium–aluminum alloy, also clad with aluminum, may also be present, as indicated in Fig. 2.

The accidents of concern are those which restrict or stop coolant flow or cause local overpower and overheating. The heat generation from the uncooled fuel can melt it rapidly, even if the reactor is immediately scrammed.

Fuel-melt phenomena have been partly characterized by a number of studies conducted at Savannah River and elsewhere. These include melting studies by Morin and Hyder,[11] in-pile transient tests in the SPERT-1 reactor,[12] and studies of fuel that reached unusually high temperatures in the high-flux charge.[13] Metallurgical studies were also made of fuel behavior near the melting point.[13] These results, when combined with studies of uranium–aluminum fuels at other laboratories,[14] have given rise to the following conclusions:

1. Localized swelling and blistering may occur in highly irradiated fuel heated above about 450°C.

2. The hot fuel expands, better contacting the ribs of the outer target and increasing heat flow through the ribs.

3. The uranium–aluminum core begins to melt at about 630°C, before the aluminum cladding melts at 660°C. This results from a lower-melting uranium–aluminum eutectic. Consequently cladding failure may release a substantial amount of the molten core before the cladding itself is completely melted.

4. Molten aluminum or core tends to flow in rivulets rather than in films. If sufficient amounts of fission-product gases are present, foams may form.[13,15]

5. Melt droplets quenched in water have an expanded "popcorn" consistency because of internal voids. As a result, they are relatively mobile in flowing water.[16]

The possible formation of metallic foams from melting irradiated fuel is important both for the manner of its relocation and for the question of the involvement of the molten fuel with the target material. The gap between fuel and target as fabricated is sufficient to keep the latter from melting at the same time as the fuel. This is an important factor in the nuclear reactivity of the degraded core. However, the expansion of the fuel as fission-product gases expand may cause good contact between fuel and target so that the two may melt together. The formation of foams is substantiated by limited SRS experience with locally overheated fuel and by unpublished studies made at Argonne National Laboratory (ANL).[15] For high burnups, these foams may have twice or more the volume of the original fuel and be stable for seconds or longer. A study by A. W. Cronenberg made under this program has predicted that fuel foaming could cause good fuel–target contact and melting of the target material at burnups of only a few percent of the fissile material.[17] Evidence for incipient foaming was also obtained in Savannah River studies of overheated fuel.[13] Hot cell experiments are currently being conducted at SRS in which coupons of irradiated SRS fuel are melted under controlled conditions to observe foam formation, swelling, and cladding failure directly.

If foaming does not occur, unirradiated (or slightly irradiated) fuel has been observed to relocate in rivulets. These may be large enough to bridge the gap between fuel and target, again leading to a complex thermal interaction with the target.

If a substantial part of the core is melted, it may spread over the bottom of the reactor vessel and flow into the primary cooling system. At this point the chemical interaction of melt with the reactor vessel and piping may be an important factor in the penetration and release of the melted material. Aluminum melts can dissolve and penetrate stainless steel at temperatures well below the melting point of the steel. Preliminary studies of this effect, as yet unpublished, were made at SRL by W. C. Mosley at temperatures ranging from 800 to 1050°C. The uranium present appears to have a significant effect in retarding dissolution. A report of this work is in preparation.

If the molten fuel penetrates the primary cooling system, it can reach the reactor building floor underneath. This would be covered by water from the moderator, the emergency cooling system, and other safety systems. Spreading of the melt over the floor will therefore depend on the balance between the heat generation of the fuel and the cooling effect of the water. The extent of spreading is important to further considerations regarding nuclear criticality and the attack of the fuel on the basemat concrete. A program of experimental characterization of melt spreading on wet and dry surfaces is in progress at Brookhaven National Laboratory (BNL) under the direction of G. A. Greene

Radioactive Source Term

Releases of noble gases, iodine, and cesium from melted Savannah River fuels were measured in a series of studies conducted prior to the SAAP (Refs. 18 and 19). These have recently been reviewed in relation to prior related studies.[20] Present knowledge may be summarized as follows.

Releases of fission products from these fuels are small until the melting point is reached. Upon melting, essentially all noble gases, much of the iodine, and significant amounts of cesium are released to the atmosphere within 2 min. At higher temperatures, the release of iodine and cesium from the fuel can be nearly complete. Under oxidizing conditions, up to 80% of the iodine present may be transported in the elemental form.

Available data do not well-define the rates of release of volatile fission products from molten fuel. Models have been developed for estimating these rates for use in the MELCOR (Ref. 21) and SCDAP/RELAP (Ref. 22) codes. An experimental program for measuring the release rates from simulated fuel melts is in progress at ANL. This study, which is part of the SAAP, is intended to provide improved release data for iodine and cesium and also to obtain information for other fission products that may be volatile at melt temperatures, such as tellurium.

Energetic Phenomena Associated with Severe Accidents

The release of radioactivity to the environment may be increased as the result of energetic phenomena that interfere with, or bypass, the normal functioning of the reactor confinement system. In addition, these phenomena can alter the course of the accident. Several such phenomena have been identified and incorporated into the severe accident program.

Molten core–concrete interactions can occur when the uncooled core contacts the concrete floor of the reactor building. These interactions have been studied in small-scale experiments at SRS and Rice University and in larger scale experiments at SNL (Ref. 23). The high mobility and chemical reactivity of the SRS fuels make the interaction quite different from its LWR counterpart Results to date have shown that at high temperatures, around 1400°C, exothermic chemical reactions occur that involve aluminum and the water of hydration in the concrete that generate hydrogen and other gases. At these temperatures, if additional water is added, the aluminum is likely to ignite. Uncooled core material heated to very high temperatures on the building floor is therefore a potentially important source of hydrogen. Because of this, molten core–concrete reactions also are potential threats to the confinement system.

Steam explosions may occur when water contacts molten core material. In these explosions the metal melt is fragmented and heat transfer to the water phase occurs within a few milliseconds. Such explosions are a concern because they can be very energetic and damaging, especially when large amounts of superheated melt are formed by a reactivity transient; this happened, for example, in the SL-1 reactor accident[24,25] and probably also at Chernobyl.[26] Such explosions are a particular concern for reactors using aluminum-based fuels because of their low melting point (about 640°C) and because of the low pressure in the primary cooling system (Higher pressures make it difficult to trigger such explosions.) Extensive investigations of steam explosions involving molten aluminum and water have been conducted within the aluminum industry and have shown that such reactions can be both energetic and somewhat unpredictable.[27,28] The unpredictability is caused by the layer of steam that forms spontaneously between the melt and the water; no explosion will occur unless this is disrupted or "triggered" The trigger may be a shock or any other effect that brings water and melt into contact. The magnitude of the explosion, once triggered, also depends on a complex variety of factors that make prediction or replication of explosions very difficult. Oxidation of aluminum, with generation of heat and hydrogen, has also been reported These factors, taken together, make accurate calculation of explosive energy and effects difficult. *Ab initio* calculations of steam explosions have been attempted, but this area remains controversial.[29,30]

The SRS severe accident program does not attempt to solve all the problems associated with steam explosions but rather is restricted to several specific issues. Two experimental studies are being conducted. Both are being done on a small laboratory scale with shock triggering to induce an explosion in a small amount of melt falling through water.[31] The first, under the direction of L S. Nelson of SNL, is aimed at investigating compositional effects on the explosion; the goal is to determine whether the addition of uranium to aluminum promotes or inhibits explosions. In this study the explosivity of aluminum and of uranium–aluminum alloys will be compared over the compositional range of interest under carefully controlled conditions. The effects of lithium addition can also be determined.

A recent addition to the program has been the study of accidents in which water flow to the control-rod assemblies is reduced. This could result in melting of some of the control rods, which are lithium–aluminum alloy. The amount of melt present at any one time is only a few hundred grams, and it is possible to do full-scale studies on melt cooling and possible steam explosions. Such studies are in progress at ANL and BNL.

In a second small-scale study of steam explosions, Prof. S. Abdel-Khalik at the Georgia Institute of Technology (Georgia Tech) is investigating the effects of dilute additives on aluminum–water–steam explosions. Additives affecting viscosity are already known to have significant effects.[32]

Nuclear criticality or "recriticality" may occur if relocation of the highly enriched fuel used in the Savannah River reactors brings it into a critical configuration. This might occur, for example, if much of the fuel, but little or no target material, collapsed into the lower part of the assembly and accumulated there. This process can, in extreme cases, produce a large, nearly instantaneous, release of radiation and heat. The result could be a steam or chemical explosion with potential for damaging the reactor vessel or building.

The nuclear reactivity of a given configuration of fissile material can be calculated. The largest number of questions on recriticality concern the rate at which criticality is approached and the process by which the nuclear reaction is terminated. These parameters determine the energy release from the recriticality. Work in this area under the SAAP work scope is presently concentrated on describing the fuel relocation process to provide the necessary rate data for recriticality evaluations. This information has been used in conjunction with steady-state recriticality analysis codes in the ongoing PRA development efforts at SRL. Additional insights related to combined neutronic and thermal–hydraulic aspects in this area are expected from the NPR severe accident program.

Hydrogen deflagration or explosion may take place if a sufficient concentration of hydrogen is formed and mixes with air. Conditions required for hydrogen to burn or explode have been characterized in the LWR severe accident program[33] and are not being investigated experimentally in the SAAP. Preliminary studies indicate that, in well-ventilated reactor buildings, hydrogen would not accumulate in flammable concentrations unless the hydrogen is generated by a process that operates on a scale of a few minutes or less. The only two such processes so far identified are molten core–concrete reactions and steam explosions, as already described. Hydrogen reaction will be considered in the analyses of these events.

Response of the Reactor Confinement System to Accident Phenomena

The response of the confinement system to challenges has been extensively studied, especially for steam and iodine:

• The response of the moisture separators and HEPA filters to steam flow was studied and reported by Peters.[34] No additional work in this area is planned as part of this program.

• A long series of reports on iodine behavior on the SRS reactor confinement system was summarized by Evans,[35] and Hyder has published additional studies.[36–38] No additional work is planned in this area under the SAAP.

• The response of the moisture separators and HEPA filters to particulate loading is being determined in studies by ANL. This work is now providing the desired quantitative information on filter blockage as a function of loading and particle size.[39,40] These studies are particularly aimed at determining the potential for pluggage of the filter system by aerosols. If the filters were plugged, building ventilation flow would cease, and the filter compartments could not be cooled by the flowing air as intended.

• Deposition of iodine and particulates within the building would reduce the amount challenging the filter compartments. A study of the effectiveness of sprays in scrubbing airborne iodine and particulates from air has been performed and published.[41,42] Sprays installed within the building could be used for this purpose.

Supporting Studies

Studies are being conducted to obtain other supporting information for use in interpreting the results of the studies described previously.

Studies of air circulation within the reactor building will be made under a contract with the South Carolina University Research and Educational Foundation.

CONCLUSIONS

The SAAP has begun to supply the basic information needed for evaluation of the behavior of the Savannah River reactors under severe accident conditions. Because of some delay in bringing the level of effort up to that intended, it is not expected to be complete in 1991, but a better understanding of the accident phenomena and system response has already been achieved in most areas. This has been particularly useful in evaluating the performance of the reactor confinement systems for restarting the reactors following the recent shutdown. Future work will be closely coupled to the risk assessment, accident management, training, and regulatory needs of the site.

Those portions of the program which deal with the behavior of molten aluminum-based fuels will be useful when evaluating the safety of other reactors that use this type of fuel. Several research reactors in the United States, including the HFIR at ORNL and the High-Flux Beam Reactor at BNL, use uranium–aluminum fuel. The most important application to another reactor will be to the proposed new heavy-water production reactor, which is now being designed. The computer codes and much of the experimental data can be used in the safety evaluation of this reactor.

ACKNOWLEDGMENT

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REFERENCES

1. *Savannah River Site Probabilistic Risk Assessment of Reactor Operations Level I Internal Events* (Report WSRC-RP-89-570) and *External Events* (Report WSRC-89-733), June 1990
2. American Nuclear Society, *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities, Boise, Idaho, October 1–4, 1990* (two volumes), Report WSRC-RP-89-1404 (CONF-9010151-6), 1990.
3. R. M. Summers, R. K. Cole, Jr., E. A. Boucheron, M. K. Carmel, S. E. Dingman, and J. E. Kelly, *Melcor 1 8 0 A Computer Code for Nuclear Reactor Severe Accident Source Term and Risk Assessment Analyses,* Report NUREG/CR-5531 (SAND-90-0364), January 1991.
4. C. M. Allison et al., *SCDAP/RELAP5/MOD2 Code Manual Volume I, RELAP5 Code Structure, System Models, and Solution Methods, and Volume II, SCDAP Code Structure, System Models, and Solution Methods,* Report NUREG/CR-5273, Vols. 1 and 2 (EGG-2555, Vols. 1 and 2), September 1989.
5. Y. M. Farawila, S. I. Abdel-Khalik, and P. J. Halverson, *K-FIX(GT) A Computer Program for Modelling the Expansion Phase of Steam Explosions Within Complex Three-Dimensional Cavities User's Manual,* Report WSRC-RP-90-1216, February 1990; *Numerical Modelling of the Expansion Phase of Steam Explosions Part I, Method and Validation,* Report WSRC-RP-90-1215, February 1990, and *Numerical Modelling of the Expansion Phase of Steam Explosions Part II, Application to the Savannah River Plant,* Report WSRC-RP-90-1214, February 1990.
6. M. T Leonard, D. L. Y. Louie, E. L. Donbroski, K. C. Wagner, M. L. Hyder, and P G Ellison, A Model for Transient Analysis of a Multiple-Medium Confinement Filter System, in *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities,* Report WSRC-MS-90-11 (CONF-9010151-5), pp. 89-98, 1990.
7 M T. Leonard, K. A Williams, and J. P. Church, Modifications to MELCOR for the Analysis of Heavy-Water Moderated, U–Al Fuel Reactors, in *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities,* Report WSRC-MS-90-9 (CONF-900917-32), pp. 104-111, 1990.
8 D. A Powers and J. E. Brockmann, Radionuclide Release and Aerosol Generation During Core Debris Interactions with Concrete, in *Proceedings of the Symposium on Chemical Phenomena Associated with Radioactivity Releases During Severe Nuclear Plant Accidents, Anaheim, California, Sept 9–12, 1986,* Report NUREG/CP-0078 (CONF-860911-Pt. 1), pp 7.1-7 36, June 1987
9. C. M. Allison, L J. Siefken, D L Hagrman, and T C. Cheng, Extension of SCDAP/RELAP5 Severe Accident Models to Non-LWR Reactor Designs, in *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities,* Report EGG-M-89534 (CONF-900917-19), pp 95-103, 1990
10 W. C Rivard and M. D. Torrey, *K-FIX A Computer Program for Transient, Two-Dimensional, Two Fluid Flow—THREED An Extension of the K-FIX Code for Three-Dimensional Calculations,* Report LA-NUREG-6623 (Suppl 2), October 1978.
11. J. P Morin and M. L. Hyder, Reaction of Molten SRP Reactor Fuel with Water, in *ANS Workshop on the Safety of Uranium–Aluminum Fueled Reactors, Idaho Falls, Idaho, March 15, 1989,* Report DP-MS-89-53 (CONF-890378-6), 1989.
12. R. W. Miller, A. H. Spano, J. Dugone, D. D. Wieland, and J E. Houghtaling, Experimental Results and Damage Effects of Destructive Test, Report on SPERT-1 Destructive Test Results, *Trans Am Nucl Soc* , 6: 138-139 (June 1963).
13 P. G. Ellison, P. R. Monson, and M. L. Hyder, Nuclear Metallic Fuel Melt Behavior During Severe Accidents, in *27th ASME/AIChE/ANS National Heat Transfer Conference, Minneapolis, Minn , July 28–31, 1991,* Report WSRC-MS-90-234 (CONF-910739-5), 1991.
14. T. Shibata, T. Tamai, and M. Hayashi, Release of Fission Products from Irradiated Aluminide Fuel at High Temperatures, *Nucl Sci Eng* , 87(4): 405-417 (August 1984).
15. L. A. Neimark, Argonne National Laboratory, private communication.
16 F. E. Brauer, N. W. Green, and R. B Mesler, Metal/Water Explosions, *Nucl Sci Eng* , 31: 551-554 (1968).
17. A. W. Cronenberg, P. G. Ellison, and M. L. Hyder, Al–U Fuel Foaming/Recriticality Considerations for Production Reactor Core-Melt Accidents, in *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities,* Report WSRC-RP-89-1422 (CONF-900917-27), pp. 400-407, 1990.

18 R E Woodley, *Release of Fission Products from Irradiated SRP Fuels at Elevated Temperature,* Report HEDL-7598, June 1986

19 R E Woodley, *Release of Fission Products from Irradiated SRP Fuels at Elevated Temperatures,* Report HEDL-7651, March 1987

20 R P Taleyarkhan, Analysis and Modeling of Fission Product Release from Heated Uranium–Aluminum Plate-Type Reactor Fuels, in *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities,* Report CONF-900917-4, pp 371-380, 1990

21 P G Ellison, P R McClure, and M T Leonard, A Parametric Model for Fission Product Release from Molten Pools of U–Al Mixtures, in *Proceedings of the International Topical Meeting on the Safety, Status and Future of Non-Commercial Reactors and Irradiation Facilities* Report WSRC-MS-90-12 (CONF-900917-34), pp 55-64, 1990

22 M J Ades, P G Ellison, and D L Hagrman, A Parametric Model for Fission Product Release from Molten Pools of U–Al Mixtures, in *Proceedings of the International Topical Meeting on the Safety, Status, and Future of Non-Commercial Reactors and Irradiation Facilities,* Report WSRC-MS-90-12 (CONF-900917-34), pp 50-54, 1990

23 H W Randolph, J E Payne, and J L Margrave, Molten Core–Granitic Concrete Reaction Studies at Savannah River, in *ANS Workshop on the Safety of Uranium–Aluminum Fueled Reactors, Idaho Falls, Idaho, March 15, 1989,* Report DP-MS-89-47 (CONF 890378-7), 1989

24 R P Taleyarkhan, Analysis of Steam Explosions in Plate-Type, Uranium–Aluminum Fuel Test Reactors, *Trans Am Nucl Soc ,* 60 421-422 (1989)

25 R O Ivins, F J Testa, and P Krause, Studies of the Aluminum–Water Reactions in TREAT, in *Argonne National Laboratory Chemical Engineering Division Summary Report, Chemical Separations Processes for Plutonium and Uranium,* Report ANL-6725, April–June 1963

26 M Ishikawa, S Shiozawa, T Wakabayashi, N Ohnishi, and H Mochizuki, An Examination of the Accident Scenario in the Chernobyl Nuclear Power Station, *Nucl Saf ,* 28(4) 448-454 (1987)

27 G Long, Explosions of Molten Aluminum in Water—Cause and Prevention, *Met Prog ,* 71 107 (1957)

28 P D Hess, R E Miller, W E Wahnsiedler, and C N Cochran, *Molten Aluminum/Water Explosions—1979,* The Aluminum Association, Washington, D C , 1979

29 T G Theofanous et al , An Assessment of Steam-Explosion-Induced Containment Failure, *Nucl Sci Eng ,* 97 259 (1987) Part I T G Theofanous, B Najafi, and E Rumble, Probabilistic Effects, p 259, Part II M A Abolfadl and T G Theofanous, Premixing Limits, p 282, Part III W H Amarasooriya and T G Theofanous, Expansion and Energy Partition, p 296, and Part IV G E Lucas, W H Amarasooriya, and T G Theofanous, Input Mechanics, p 316

30 See, for example, the spirited exchange of viewpoints beginning on p 149 of the October 1988 issue of *Nucl Sci Eng* 100(2)

31 L S Nelson, T Fuketa, M J Eatough, and F J Vigil, The Triggering of Steam Explosions of Single Drops of Pure and Alloyed Molten Aluminum, *Trans Am Nucl Soc* 61 458 (1990)

32 L S Nelson and K P Guay, Suppression of Steam Explosions in Tin and Fe-Al_2O_3 Melts by Increasing the Viscosity of the Coolant, *High Temp -High Pressures* 18(1) 107-111 (1986)

33 M Berman, A Critical-Review of Recent Large-Scale Experiments on Hydrogen–Air Detonations, *Nucl Sci Eng ,* 93(4) 321-347 (1986)

34 A H Peters, *Application of Moisture Separators and Particulate Filters in Reactor Containment* Report DP-812, December 1962

35 A G Evans, *Confinement of Airborne Radioactivity Final Progress Report January/December 1978,* Report DP-1542, February 1980

36 M L Hyder, Mechanisms of Carbon Aging and Their Effects on the Retention of Organic Iodides by Carbon, in *Gaseous Effluent Treatment in Nuclear Installations, Proceedings of a European Conference Held in Luxembourg on 14–18 October, 1985,* G Fraser and F Luykx (Eds), Report CONF-851082-, pp 451-462, Graham & Trotman, London, 1986

37 M L Hyder, Effects of Organic Material on Carbon Aging, in *Proceedings of the 20th DOE/NRC Nuclear Air Cleaning Conference, Boston, Mass , August 22–25, 1988,* Report NUREG/CP-0098-Vol 1 (CONF-880815), May 1989

38 M L Hyder and R A Malstrom, Factors Affecting the Retention of Methyl Iodide by Iodide-Impregnated Carbon, in *Proceedings of the 21st DOE/NRC Nuclear Air Cleaning Conference, San Diego California Aug 13–16, 1990,* Report NUREG/CP-0116-Vol 2 (CONF-900809-Vol 2), 1991

39 V J Novick, P J Higgins, B Dierkscheide, C Abrahamson, W B Richardson, P R Monson, and P G Ellison, Efficiency and Mass Loading Characteristics of a Typical HEPA Filter Media Material, in *Proceedings of the 21st DOE/NRC Nuclear Air Cleaning Conference, San Diego, California, Aug 13–16, 1990,* Report NUREG/CP-0116-Vol 2 (CONF-900809-Vol 2), 1991

40 P R Monson, P G Ellison, and V J Novick, Experimental Relationship Between the Specific Resistance of a HEPA Filter and Particle Diameters of Different Aerosol Materials, Presented at the 6*th International Symposium on Heat and Mass Transfer, Miami, Florida, Dec 10–12, 1990* Report WSRC-MS-90-257 (CONF-901228-1), 1990

41 M L Hyder, Experimental Measurements of Air Cleaning with Water Sprays, in *Proceedings of the 20th DOE/NRC Nuclear Air Cleaning Conference, Boston Mass , August 22–25, 1988,* Report NUREG/CP-0098-Vol 2 (CONF-880815-Vol 2), pp 650-667, May 1989

42 M L Hyder, Measurements of Iodine Uptake in Falling Water Droplets, *Nucl Technol ,* 94(1) 80-86 (1991)

Control and Instrumentation

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A Framework for Selecting Suitable Control Technologies for Nuclear Power Plant Systems

By R. A. Kisner[a]

Abstract: *New concepts continue to emerge for controlling systems, subsystems, and components and for monitoring parameters, characteristics, and vital signs in nuclear power plants The steady stream of new control theories and the evolving state of control software exacerbates the difficulty of selecting the most appropriate control technology for nuclear power-plant systems As plant control room operators increase their reliance on computerized systems, the integration of monitoring, diagnostic, and control functions into a uniform and understandable environment becomes imperative A systematic framework for comparing and evaluating the overall usefulness of control techniques is needed This article describes nine factors that may be used to evaluate alternative control concepts These factors relate to a control system's potential effectiveness within the context of the overall environment, including both human and machine components Although not an in-depth study, this article serves to outline an evaluation framework based on several measures of utility*

An effort to develop advanced control and information systems is under way by commercial entities, academic institutions, and national laboratories.[1] Such an effort is under way at the Oak Ridge National Laboratory (ORNL) as a part of the Advanced Control Program funded by the Department of Energy (DOE). ORNL has developed and evaluated several advanced control methods and systems.[2–13] The purposes of an advanced control system are to bring about improved performance, reliability, and maintainability both for the plant and the control equipment. In many cases advanced control systems directly help the plant operator by providing automated operation and concise information about plant status.

Besides functioning to maintain control of the processes to which it is assigned, a control system must interface and interact with other distinct environments, including plant maintenance and operations. Because of these interactions, selection of control techniques and implementation of the control algorithms should be performed in consideration of human and plant interactions. This stands in contrast with developing a control algorithm as an isolated task.

Previous studies[14] have concentrated largely on measures of dynamic performance but have neglected other equally important considerations. The identification and definition of additional measures of merit are the objectives of this paper.

A control system's merit can be evaluated by applying a series of tests that measure its utility.[15,16] A system's utility is an expression of its suitability for the intended mission, which includes the notions of effectiveness, practicality, compatibility, and serviceability. *Measures of utility* are quantitative and qualitative criteria that express how well the system meets the mission requirements. The measures include the traditional quantitative performance criteria (e.g., specific response to perturbation) as well as qualitative mission-related factors. Such factors include reliability and availability of output

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generated, robustness (e.g., tolerance to changes in the original characteristics of the plant or other environments), requirements placed on computational and other resources, treatment of downstream components and systems, and knowledge and training requirements placed on the operating crew and other plant personnel.

This article presents some of the issues that should be considered during the selection of control technologies and the design of control functions and algorithms. Nine factors are presented that can be used by a system designer to develop methods for measuring the usefulness of a control technology. The methods that result may be used to assess a control system's potential effectiveness within the context of the overall environment, including both human and machine components.

BACKGROUND

The nuclear power-plant control room is evolving into a more integrated and automated management center for support of the operator and operation of the plant. To a large extent, the efforts of many developers across the United States and internationally are uncoordinated. Since designers of controllers and control room equipment have not found a universal control technology or system organization, engineering judgment and trade studies remain part of the design process.

New control algorithms are emerging as analog electronic controllers are replaced with digital controllers in existing plants. The capabilities of software-based systems to perform more complex calculations and logical operations will clear the way for far more effective control. However, early digital replacements were programmed as direct functional replacements, including faithful emulation of the original control law. Developmental work by DOE has shown that improved plant performance is possible with advanced control algorithms.[17,18]

When considering control system development, some generalizations can be made. Real-time control for power plants falls under two basic types: (1) continuous system and (2) discrete event. A continuous system exhibits a continuous variation of states so that variables can be proportionally controlled. In contrast, a discrete event system may assume only discrete operational states (e.g., *off* or *on*) and thus may not be controlled proportionally. An automated control system merges both control types to effect multiple modes of operation and achieve coordination of multiple subsystems. Among the methods of control for continuous systems, the current breadth and diversity of research and development (R&D) on controller types and design methods at ORNL are shown in Table 1.

MEASURES OF UTILITY FOR CONTROL SYSTEMS

A control system designer must choose the most appropriate control technique for each of the processes to be controlled. However, determining what is appropriate may involve broader issues than a traditional performance assessment, such as time- and frequency-domain response. Other factors may become important when comparing dissimilar control architectures, especially when the control system's sphere of interaction includes other machine and human systems. The designer should realize that there are additional measures of utility by which to compare and select a control technique. These additional measures of utility can be used as the basis of detailed cost–performance analyses for objective evaluation of control technology and strategy choices.

Systematic analysis of the properties of control systems is possible by evaluating and ranking their measures of utility. These measures are especially useful for comparing alternative or competing equipment or software designs. Nine categories are proposed for evaluating control systems. Although not exhaustive, this list represents a plausible means of cross-comparing the benefits of similar equipment and designs. Measures of utility for evaluation and comparison of control systems are shown in Table 2. No attempt has been made in this article to rank the importance of each measure relative to each other since the significance and relevance of each measure depends on the specific control system application. The summary discussions that follow describe each measure and consider means of assessment.

Compatibility with Human Operators

Good design practice dictates that up-front analyses (e.g., user analysis, task analysis, and allocation of functions) be performed to help determine the assignment of control functions to machine systems.[19] Allocation of control-related functions forms the basis for establishing the operating staff's role. Once these allocations are made, issues of operator compatibility can be addressed.

Meaningfulness of Information. Information presented to the operator must be unambiguous and appropriate for the tasks to be performed. The information format should organize and underscore meaningful information for the human. The interface system must be

Table 1 Controller Types and Design Methods Currently Under Study or in Use at ORNL

Controller or design method	Use of mathematical process model	Recent ORNL activities (with references)	Comments
Proportional-integral-derivative (PID) feedback and feedforward	Model used only for simulation testing	Developed improved integral windup prevention algorithms Nonlinear gain compensators applied to valve control Combined feedforward and feedback to control reactor and feedwater systems[4 11 27]	Commonly used in plant control Operators very familiar with PI loops Can be made multivariate to small degree Introduction of feedforward allows reduction of closed-loop gains for stability
Linear quadratic Gaussian (LQG) regulator	Model used in the calculation of feedback gains and as an estimator-filter for calculating unmeasurable variables	Experimented with robust design techniques, compensation for time delay in multiple-input–multiple-output systems, gain scheduling, and inclusion of feedforward signals [3–7 17 24,27]	Successful applications in balance-of-plant systems Lack of robustness is an issue
Nonlinear optimal control with compensation for unmodeled dynamics	Model used to formulate the state-space equations and adjoints	Nonlinear state-space controllers constructed through nonlinearized application of Maximum Principle [2 9 10]	Technique applicable to numerous reactor subsystems Robust and inherently adaptive
Adaptive control	Model is active within the controller, adapting gains to changing or nonlinear conditions	Evaluating adaptive feedforward and self-tuning regulator concepts[5 6]	Operators want authority over a controller's adaptation
Fuzzy-logic control	Model used in simulation testing	Developed hierarchical application of fuzzy rules [5 6,8 12]	Direct hardware implementation of fuzzy rules has potential benefits
Reconstructive control [inverse dynamics]	Model used directly to form feedback compensator	Applied reconstructive dynamics technique to several nuclear power systems[5 6 18]	Technique developed at ORNL shows good results on nonlinear systems
Learning algorithms	Learning algorithms use model to iterate on gains and offsets	A generalized learning algorithm was developed for an application [13]	Another application employed the 2-D Roesser Equation to startup of EBR-II reactor
Neural networks	Model used to train network	Applied neural network to power plant start-up[5 6]	The governing rules are implied in the nodes More work has been done in signal analysis than control
Predictive control	Model directly predicts future conditions based on current control actions	Developing object-oriented simulation methods	A computationally intensive technique
Expert systems	Model used to test expert system	Developed expert systems for effecting supervisory control [9 10]	Rules are explicit Can be combined with other control methods

able to integrate items so that they are consistently meaningful.[20] However, the effectiveness of control system–operator interaction depends not only on the interface design but also on the control techniques and strategies employed and the degree of automation. In most instances, reporting meaningful information to the operator is necessary to maintain confidence in the controller and to maintain operator awareness of the system state in case human intervention is required. More information is not necessarily better, however. Although a quantitative measure for information reporting does not exist, the reporting aspect should be considered in evaluating control algorithms and their means of implementation. The following three information categories pertain to controller design:

Explanation of controller actions. An operator's trust in a controller is not solely a function of its reliability. Among the factors that contribute to user acceptance is a

Table 2 Measures of Utility for Control Systems

1 Compatibility with human operations • Meaningfulness of information Explanation of controller actions Observation of unmeasurable states Tracking of plant parameters • Understandability Complexity Match to training and education	5 Interactions with nearby components and subsystems • Actuators • Subsystems 6 Ability to tune in the field • Ability to verify controller tuning • Complexity of tuning process • Disruption to the process
2 Real-time quantitative performance and stability • Dynamic performance • Frequency-domain characteristics • Static performance (accuracy and precision of results)	7 Resource requirements • Real-time computational requirements • Sensor count, accuracy, and bandwidth requirements • Communication network requirements
3 Reliability of results or conclusions • Opportunity for branching to incorrect path • Repeatability of decision	8 Development considerations • Design resources and effort • Verification, validation, and testing
4 Tolerance to degraded conditions and robustness • Modeling errors • Noise corruption • Process Parameter Variation • Sensor and actuator failure	9 Long-term considerations • Flexibility–upgradability • Maintainability • Compatibility

controller's ability to explain its actions. Two important facets of explanatory information are (1) what the controller is doing now and why and (2) what the controller is about to do next, when its next action is to occur, and why that action is planned (if asked). The importance of the latter information should be stressed. The following account of experience with an automated aircraft landing system illustrates the point:[a]

> Some years ago, an aircraft landing system was developed that automated landing from the point at which the aircraft passed the outer marker, through the inner marker, to touchdown. The controller signaled the pilot at each checkpoint, indicating when it found the outer and inner markers, respectively. However, at the point of flare (transition from steep descent to parallel)—the most critical phase of landing—the controller gave indication only at the moment of initiation. The pilot monitored progress through the checkpoints; however, during the last 50 ft of descent, he became concerned that the controller might fail to flare. Inevitably, the pilot intervened to manually flare the craft. This occurred in almost all uses of the system because the controller did not indicate what it was intending to do and when.

Observation of unmeasurable states Some model-based controllers may employ an active plant model as an observer–filter to estimate unmeasurable states for full-state feedback. Depending on how the plant is modeled, these states may provide useful information to the operator about unmeasured parameters, for example, fuel temperature and a multitude of other unmeasurable temperatures, xenon concentration, and heat fluxes.

Tracking plant parameters. Component parameters, such as tank volumes and heat-transfer coefficients, may change as the plant ages. Model-based controllers, which periodically update their internal plant model, may be used to detect deviations from design-basis specifications. This tracking capability can prove beneficial for trend analysis of deposit buildup, fouling, leakage, wear, and other component degeneration.

Understandability. Information presented to the operator must be simple, clear, and understandable. The structure, format, and content of display dialogue must result in effective communication. However, in the context of a control system, user understandability goes beyond structure, format, and content of display. A criticism often heard in utility circles concerns the lack of predictability of controllers during abnormal situations, especially transients. Operators, sometimes as a matter of practice, revert to manual control at the onset of a transient. As explained earlier, part of the problem relates to lack of explanation by the controller as to what it intends to do next. Another part of the problem is the operator's lack of understanding or his misunderstanding of how

[a]From a personal communication with H E (Smoke) Price, 1988

the control strategy or technique actually works. The potential is high for occasions in which the operator seizes manual control, even though the controller might have chosen the correct course of action.

Complexity. Even with training, education, procedures, and system drawings, operators consider the controller untrustworthy during unexpected situations. Complex controllers should not be designed to be black boxes. Operations personnel should be required to communicate with and understand the functionality of the controller without being required to understand complex mathematics or become computer programmers.[21] Complexity beyond that needed to undertake the control strategies or meet the system's goals must be considered excessive and may lead to lack of trust.

Match to operator training and education. Current operator and maintenance technician training is focused on comprehending single-input, single-output (SISO) analog controllers, which even nonscientific personnel find relatively easy to understand. However, as control technology becomes software-based, additional technical training is required for personnel to fully comprehend the internal functions and decision-making processes of advanced control systems. This is especially true of multivariate controllers, which may perform control actions that seem counter-intuitive at the time.

Real-Time Quantitative Performance and Stability

Dynamic Performance. Time-domain measures of performance are well known[22] for SISO systems. Control system dynamic performance can be evaluated by measuring rise time, overshoot, settling time, and integral square error (or other quantitative error criteria). A rule-of-thumb classical transient response criterion states that a controlled system should exhibit less than 30% overshoot to a step change in set point. Other criteria are system dependent and must be determined from plant or system requirements. These criteria may not have the same meaning or usefulness for fuzzy-logic, neural network, or expert system controllers.

Frequency-Domain Characteristics. Frequency-domain measures of performance are well understood for SISO systems[23]; however, for state-variable multiple-input, multiple-output (MIMO) systems, the concept of singular-value analysis must be used. In general, graphical techniques are the mainstay of frequency-domain performance analysis. Examples include Bode plot, Nyquist plot, root-locus, and others. From the Bode plot, stability measures, such as gain and phase margin, are calculated. For SISO systems, a 45° phase margin and a 6- to 12-dB gain margin are desirable. Frequency-domain singular-value analysis is discussed further as part of the robustness topic.

Static Performance (Accuracy and Precision of Results). Steady-state performance of the controlled system is measurable in terms of closeness to desired value, repeatability, and lack of hunting. For systems that require zero steady-state error, an integral control action is often employed; in other cases, a small error is acceptable. In any case, steady-state operation should be tested against a specific requirement. Testing may also be performed to determine whether the control system's response is repeatable, that is, whether it can return the system to the desired end point from different starting points. Static friction as well as other nonlinear phenomena can contribute to nonrepeatable behavior.

Reliability of Results or Conclusions

In keeping with the theme of this article, this discussion of reliability is focused specifically on issues related to the theories or methods behind the algorithms used in a control system. A discussion of the overall control system reliability would necessarily include concepts of fault-tolerant design approaches and hardware considerations, which is beyond the scope of this article.

One issue concerns unique errors that can potentially arise in algorithms that incorporate decision points or employ branching logic, as would be the case in an expert system, for example. In this context, the notion of reliability is distinct from that of system stability, which is elaborated in the previous section. Control systems that use rule-based logic can arrive at improper conclusions concerning system status or the proper rule to apply for a particular condition. It is therefore possible for a control system that allows for alternative conclusions through "branching" to diverge onto a path that is incorrect for the mode of operation. Improper operation of the controlled process results if a control system acts on such incorrect conclusions. Further, because of the number of and complexity of rules in a practical implementation, similar circumstances may not result in the same conclusions; hence the results are not entirely repeatable. Organizing rules into groups according to a priority scheme can place limits on how far the branching can stray.

User and operator confidence that a control system will make correct decisions at significant moments is important to maintain. To maintain such confidence, intelligent controllers must produce a high percentage of credible conclusions and must be scrutable. The hardware and software combination must exhibit reliability exceeding both the requirements of the systems being controlled and the expectations of the operators.

Tolerance to Degraded Conditions and Robustness

Modeling Errors. The classic definition of a robust control system is one that is insensitive (i.e., maintains performance and stability) to bounded plant parameter variations, disturbances, noise, and high-frequency plant perturbations. The theoretical problem addressed by robustness is the design of accurate control systems given plants that contain significant uncertainties, some of which result from modeling errors. Because of imperfections in the mathematical and conceptual models of systems, some dynamics are inevitably neglected. Even though some dynamical effects are benign and may be neglected during design, the final design must exhibit robustness to compensate for unmodeled dynamics and meet the specification.

One approach to robust control system design being explored is the linear quadratic Gaussian (LQG) technique with loop transfer recovery (LQG/LTR) (Ref. 24). This technique, which can be applied to MIMO or SISO systems, uses frequency-domain design techniques to determine the full-state feedback and Kalman filter gains. The technique seeks a balance between robustness and command-following (i.e., rise time). The literature[25] reports other techniques for solving the robust control problem.

Noise Corruption. All systems will experience noise, which may be introduced by the process or by the sensors and their associated electronics. Noise may be modeled as a random signal added linearly to the control and measurement signals. Robustness to additive noise may be examined through mathematical analysis or simulation. Through mathematical analysis, the high-frequency attenuation of the maximum singular value for the combined linear controller-plant matrix can be calculated and plotted against frequency. For power-plant control, the maximum singular value should exhibit small magnitudes for frequencies beyond a few radians per second; for aircraft design (as a comparison), it should fall off rapidly beyond 5 to 10 radians/s. By limiting the high-frequency response of the system through controller design, sensitivity to high-frequency noise disturbances can be minimized.

Simulation testing allows evaluation of noise effects by directly inserting noise into the system. This test can be applied to both linear and nonlinear systems. A controller's response to noise should not cause instability or obscure the true signals. A controller should attenuate the noise at the output to a level that does not cause excessive wear in downstream components. Adaptive controllers should not incorrectly adjust gains because of deleterious noise.

Robustness to noise can be specified by the maximum magnitude of noise (over a frequency bandwidth) that can be tolerated by the system under test. A Gaussian noise process is usually assumed, although other noise sources can be used, such as a spike generator.

Process Parameter Variation. Control systems should exhibit reasonable robustness against parameter changes in the actual process. This is especially true of controllers that utilize a mathematical model either during design or as an active part of the controller. Modeling approximations and mismatches introduce time-variant errors into the control signals, which can lead to performance degradation or instability unless external correction is provided. Component parameter variation can be sudden or progressive in development owing to blockage, buildup, or other causes.

For robustness for fixed-gain controllers, a filter and feedback gains are chosen so as to reduce sensitivity to parameter deviations. Time response may have to be sacrificed to reach the insensitivity to parameter deviation required for some applications. For adaptive (or self-tuning) controllers, the gains are adjusted by an adaptation mechanism and the internal model is maintained current.[26] Analytical assessment of robustness to process parameter variation is possible for linear systems by examining the minimum singular value for the controller–plant combination. Disturbance rejection and insensitivity to parameter variations are improved by higher gains at low frequencies. This is analogous to gain margin in SISO systems.

Simulation of a nonlinear model of the plant is recommended for parameter robustness testing. This entails calculating the worst-case deviation of plant parameters on the basis of the physics of the components involved. Time-domain performance is analyzed by introducing abnormal plant parameters.

One plant parameter of special note that affects robustness is process and measurement time delay. Although the controller may compensate for typical fixed delays found in the plant and the sensors, the controller may not tolerate the maximum delay possible under worst-case conditions. This could result in the closed-loop system becoming unstable. A recent study[27] has shown that a control system that may be robust to parameter variations, disturbances, noise, and model uncertainty may not necessarily be closed-loop stable when significant process delay is present. At this time, no formal analysis method is available; thus evaluation of delay tolerance requires simulation.

Sensor and Actuator Failure. A comprehensive signal validation research program for advanced reactors[28] is currently under way at The University of Tennessee and at ORNL. Understanding the controller-plant response upon loss of sensor signal remains an important issue because not all controllers may include a front-end signal validator. Additionally, the failure of the validator must be considered. Except for redundancy of valves, motors, and heaters, no means of overcoming failed actuators may be possible.

Testing for loss of sensor data and actuator control should be performed with a simulation environment based on nonlinear component models. Analysis of the failure modes of sensors and actuators provides the basis for the events to be introduced into the simulation. Some hardware implementations of fuzzy-logic algorithms possess the intrinsic quality of graceful degradation under environmental stress.[29] This quality may be useful in sensor, actuator, or control electronics and should be considered when comparing algorithmic techniques.

Interactions with Nearby Components and Subsystems

Actuators. Actuation devices (e.g., valves, relays, and motors) that receive commands from a controller may be subject to wear-out or other failure phenomena that can be accelerated or retarded by the action of the controller. A controller that continually generates *excessive* control actions may induce premature actuator failure compared with one that generates *minimal* control actions. An algorithm's method of achieving control signals may affect actuator lifetime and maintenance requirements. Usually, these effects are cumulative rather than instantaneous.

Although the primary objective of a control system is to govern process performance, a secondary objective may be to preserve actuator integrity and limit stress cycles on process components. Considering an actuator's stress factors—mechanical, thermal, electrical, chemical, and radiation—only mechanical and electrical factors are under the direct influence of a controller's actions. Typical contributors to excessive control action are noise in the signal path, excessive gain, and control loop oscillations. Time-honored means of prevention of excessive control are use of deadband and low-pass filtering.

A simple method to measure valve stress would be to form a quadratic cost index in which the square of valve motion is integrated. A further enhancement of the formula might be to include a factor for thermal stress. The valve stress factors for several controller designs can be compared under both steady-state (with typical noise) and transient conditions with simulation techniques.

Subsystems. In a fashion similar to the actuator receiving direct commands from a controller, consideration should be given to the manner in which a controller may affect downstream and upstream subsystems, which may include other controllers. Generally, hierarchically organized control systems address well the problem of disturbance propagation from subsystem to subsystem.

Ability to Tune in the Field

A long history of tuning proportional-derivative-integral (PID) controllers has established a pattern and convention for loop gain adjustment.[30] Because of this, the capability to verify controller tuning and to actually perform tuning in the field can be taken as a requirement for nuclear power-plant operations. Controllers are required to adapt to slowly varying component behavior because long-term changes in plant conditions must be expected. A good objective might be to design robust and adaptive controllers that minimize the amount of field tuning required; however, the option of field verification and adjustment should continue to be provided.

Tuning methods for multivariate controllers are different from those for SISO controllers. The procedure for manual tuning of SISO PID loops is applied to each individual loop of a system. Translation between tuning parameters and controller gains is minimal, which gives the person performing the tuning a feel for the gain's effect on system performance. In contrast, tuning an LQG/LTR controller represents a design-level effort. The complexity of the procedure greatly increases over that which is required for the single-loop case. Individual gains in the feedback matrix cannot be adjusted independently because of the interactive nature of multivariate control. Several layers of mathematical transformation are required between statement of the performance objective and determination of the gains. The LQG/LTR tuning parameters are expressed in terms of sensitivity and rejection margins, which is different from the SISO controller. The complexity of the mathematics requires a computer for the calculations. This implies a need to embed some minimum level of control system development tools in the field controller. In this way, field tuning can be considered a microcosm of the controller design process.

More R&D is needed to provide useful and acceptable tuning capabilities for advanced control techniques. Reevaluation of the tuning problem, considering the capabilities of microcomputers, may reveal an improved scheme for field tuning and an improved set of tuning parameters than are now in use, including automatic tuning schemes with report generation.

Resource Requirements

Several resources are affected by the choice of control strategy and control technique. A short discussion follows.

Real-Time Computational Requirements. Generally, model-based control techniques are the most demanding on computer processing speed and memory. Current generation digital controllers offer quite significant computing power; however, some techniques present a challenge to this level of computer technology. Faster-than-real-time simulation, used as a control technique, poses a great computational burden that may require several more development generations of computer hardware for effective implementation. Increased computational power incurs higher cost and may decrease reliability. Computational burden is a consideration in comparing control techniques.

Sensor Count, Accuracy, and Bandwidth Requirements. The number of sensors (especially critical signals) needed for control and their accuracy requirements vary, depending on the strategy and control technique chosen. One criterion for selecting a controller may be to find the one that requires the minimum number of inputs to accomplish the control objective. However, this may not be an entirely realistic criterion because it does not consider which sensor is required. Some sensors are more expensive than others, some are more prone to failure, and some may be chosen already (e.g., in a retrofit installation). Also, some controller designs may force stringent accuracy or noise requirements on some measurements, and this aspect must therefore be considered in evaluating controller strategies and techniques. Keep in mind that the number of sensors can exceed the number of inputs because of redundancy.

Communication Network Requirements. The data highway that interconnects controllers has a significant effect on the performance of the overall control system. The recent failure of CEGELEC's ControBloc P20 control system communication network to pass inter-process data without interruption shows the importance of data highway speed and reliability.[31] Some distributed hierarchical control architectures require large amounts of inter-module communications; thus heavy demands are placed on the network resource. The choice of control technique and strategy may therefore be limited by the performance and availability of a high-speed data highway.

Development Considerations

Design Resources and Effort. Since the technical expertise required of the algorithm design team varies widely, depending on the control technology chosen, some technologies may be beyond the resource scope of some organizations to develop and implement. This limitation applies as well to development tools and computing resources. Recent developments in computer-aided control system design (CACSD) environments have made feasible rapid control algorithm development for many emerging techniques.[32] Many of these development tools also support a simulation environment.

Verification, Validation, and Testing. Nuclear industry conservatism and requirements of the Nuclear Regulatory Commission (NRC) and other regulatory bodies impose a high degree of analysis and quality checking on the design and implementation of the instrumentation and controls (I&C) that go into a nuclear power plant. Many rigorous standards and guides apply to the reactor protection system and other safety-related equipment; however, fewer NRC regulations apply to control systems. Although the full orb of software verification and validation (V&V) and testing may not be necessary for control and information systems, reliability and cost effectiveness may be enhanced through the application of V&V. This being the case, the choice of control algorithm and the entire control system should balance ease of performing V&V and testing with other factors.

Long-Term Considerations

Flexibility–Upgradability. Digital control systems inherently offer a high degree of flexibility for modification of parameters and algorithms, and systems designers should plan on future upgrades. The control software package (including the algorithm) should offer capability for future expansion with minimal detrimental side effects.

Maintainability. The software and hardware package should be field maintainable since coding errors and other problems are sometimes discovered only during real plant operation. Also, as equipment is replaced in the course of time, software may need maintenance to function on the replacement hardware

Ironically, the *time-to-obsolescence* of computer hardware may be shorter than the analog equipment being replaced. The time span from introduction to obsolescence for discrete analog and logic circuit components may extend to 25 years By contrast however, microprocessor and memory components have a span of typically 10 to 15 years This shorter computer product lifespan underlines the need for creating long-term standards for manufacturing I&C systems Such a standardized environment would reduce the risk that equipment developed at some future time may be incompatible with earlier generations

Compatibility Increasingly, digital-computer-based equipment will replace analog electronic-based equipment as electric utilities seek to ensure continued power-plant operation beyond original design life Many equipment suppliers will be providing hardware and software as upgrades continue However, digital computer equipment from different vendors may be incompatible in many ways Each vendor's I&C equipment has unique features, proprietary communications protocols, non-interchangeable software, and distinctive hardware configurations. The compatibility issue should be considered when selecting control system technologies

CONCLUSIONS

The function and design of nuclear power-plant control systems is evolving owing to the flood of new ideas, software, and equipment, the potential improvements these offer, and other factors, such as the need to reduce operator error and an emphasis on reducing maintenance costs. Systematic evaluation of various alternative control system design concepts must be performed to help designers integrate the concepts for maximum benefit. At best, such integration and coordination will be a compromise The measures of utility described here and the surrounding issues provide an initial framework for comparing control system properties. Further work is necessary to expand and refine the measures of utility so as to create a comprehensive testing procedure for systematic and objective evaluation of control technology and strategy choices The author is aware that a case study would clarify application of the measures. A benchmark problem is being developed that can be used as a test for evaluating control systems according to the measures of utility described.

REFERENCES

1 James D White, David D Lanning Co-Chairmen, Leo Beltracchi Fred R Best, James R Easter, Lester C Oakes, and A L Sudduth, *WTEC Panel Report on European Nuclear Instrumentation and Controls* Report PB92-100197, December 1991

2 C March-Leuba, *Study and Development of Advanced Control Techniques for Nuclear Reactors and Robots*, Report ORNL/TM 11238 August 1989

3 J M Bailey, B R Upadhyaya, and R E Battle, *Optimum Control Techniques for Nuclear Power Plants*, Report ORNL/TM 10372, October 1987

4 R C Berkan, B R Upadhyaya, and R A Kisner, *Implementation of Multivariable Control Techniques with Application to Experimental Breeder Reactor II*, Report ORNL/TM-11134 June 1990

5 R C Berkan, B R Upadhyaya, R L Bywater, and R A Kisner, *Advanced Automation Concepts Applied to Experimental Breeder Reactor II Startup* Report ORNL/TM-11716, August 1991

6 R C Berkan, B R Upadhyaya, L Tsoukalas, R A Kisner, and R L Bywater, Advanced Automation Concepts for Large-Scale Systems, *IEEE Control Systems* 11(6) 4-12 (October 1991)

7 R A Kisner and G V S Raju, *Automating Large Scale Power Plant Systems A Perspective and Philosophy* Report ORNL/TM 9500, December 1984

8 G V S Raju, J Zhou, and R A Kisner, Hierarchical Fuzzy Control *Int J Control* 54(5) 1201 1216 (November 1991)

9 P J Otaduy, C R Brittain L A Rovere, and N B Gove, A Distributed Hierarchical Architecture of Expert Systems for Supervisory Control of Multimodular Nuclear Reactors, *Proceedings of the International Conference on Frontiers in Innovative Computing for the Nuclear Industry* Report CONF-9109110-5, 1991

10 P J Otaduy, C R Brittain, L A Rovere, and N B Gove, Supervisory Control Conceptual Design and Testing in ORNL's Advanced Controls Research Facility, *Proceedings of the International Conference on Frontiers in Innovative Computing for the Nuclear Industry*, Report CONF-9109110-1, 1991

11 C March-Leuba, J March-Leuba, and T L Wilson, A New Approach to Controlling the Water Level of U Tube Steam Generators, *Nuclear Science Symposium* Report CONF 900143-3, 1989

12 L H Tsoukalas, R C Berkan, B R Upadhyaya, and R E Uhrig, Expert Systems Driven Fuzzy Control Application to Power Reactors, *Proceedings of the Second International Conference on Tools for Artificial Intelligence Washington DC November 6 9, 1990*

13 C R Brittain, P J Otaduy, and R B Perez, *Development of a General Learning Algorithm with Applications in Nuclear Systems* Report ORNL/TM-10650, December 1989

14 C M Krishna and K G Shin, Performance Measures for Control Computers, *IEEE Trans Autom Control* AC-32(6) 467 473 (June 1987)

15 R A Kisner, R J Carter, and R W Lindsay, Issues of Integrating High-Tech Concepts into Nuclear Power Plant Operation, in

Proceedıngs of the ANS Topıcal Meetıng on Advances ın Human Factors Research on Man\Computer Interactıons Nuclear and Beyond, Report CONF 900607-5, pp 295-303, 1990
16 R A Kısner, R C Berkan, and B R Upadhyaya, Performance Charactenstıcs for Advanced Control Systems, *Proceedıngs of the Seventh Power Plant Dynamıcs Control, and Testıng Symposıum,* Report CONF-890555-12, 1989
17 T L Wılson and W K Wagner, Multıvanable Control for the PRISM ALMR, *Proceedıngs of the Seventh Power Plant Dynamıcs, Control and Testıng Symposıum,* Report CONF-890555-14, 1989
18 R C Berkan, B R Upadhyaya, L H Tsoukalas, and R A Kısner, Reconstructıve Inverse Dynamıcs Control and Applıcatıon to Xenon-Induced Power Oscıllatıons ın Pressunzed Water Reactors, *Nucl Scı Eng*, 109(2) 188-199 (1991)
19 R Pullıam, H E Prıce, J Bongarra, C R Sawyer, and R A Kısner, *A Methodology for Allocatıng Nuclear Power Plant Control Functıons to Human or Automatıc Control,* Report NUREG/CR-3331 (ORNL/TM-8781), August 1983
20 D I Gertman, H S Blackman, W W Banks, and R J Peterson, *CRT Dısplay Evaluatıon The Multıdımensıonal Ratıng of CRT-Generated Dısplays,* Report NUREG/CR-2942 (EGG-2220), 1982
21 R A Kısner, T B Shendan, and B R Upadhyaya, Computıng and Cognıtıon ın Future Power Plant Operatıons, ın *Proceedıngs of the Fıfth Power Plant Dynamıcs Control, and Testıng Symposıum,* Report CONF-830318-, pp 25 01-25 11, 1983
22 J J D'Azzo and C H Houpıs, *Lınear Control System Analysıs and Desıgn,* McGraw Hıll, New York, 1981
23 B Fnedland, *Control System Desıgn* McGraw-Hıll, New York, 1986
24 G V Murphy and J M Baıley, *Robust Control Technıque for Nuclear Power Plants,* Report ORNL/TM-10916, March 1989
25 P Dorato (Ed), *Robust Control,* IEEE Press, New York, 1987
26 V V Chalam, *Adaptıve Control Systems Technıques and Applıcatıons* Marcel Dekker, Inc , New York, 1987
27 G V Murphy and J M Baıley, Evaluatıon of Tıme Delay Requırements for Closed Loop Stabılıty Usıng Classıcal and Modern Methods, *Proceedıngs of the 21st Southeastern Symposıum on System Theory,* Report CONF 890361-1, 1989
28 B R Upadhyaya, K E Holbert, and T W Kerlın, Development of an Integrated Sıgnal Valıdatıon System and Applıcatıon to Operatıng Power Plants, ın *Proceedıngs of the Seventh Power Plant Dynamıcs Control and Testıng Symposıum* Report DOE/NE/37959-28 (CONF-890555-10), 1989
29 T Yamakawa, Intrınsıc Fuzzy Electronıc Cırcuıts for Sıxth Generatıon Computıng, ın *Fuzzy Computıng,* M M Gupta and T Yamakawa (Eds), Elsevıer Scıence Publıshers B V , Amsterdam, 1988
30 B G Lıptak and K Venczel (Eds), *Instrument Engıneers Handbook—Process Control,* Chılton, Radnor, Pa , 1985
31 A MacLachlan, P Marshall, and E Hıruo, Westınghouse Supplants CEGELEC as the Sızewell-B I&C Supplıer, *Nucleonıcs Week* 32(24) 1 (June 13, 1991)
32 M Jamshıdı, M Tarokh, and B Shafaı, *Computer-Aıded Desıgn of Modem Control Systems,* Prentıce-Hall, 1989

WORKSHOP ON PC-PRAISE: A PROBABILISTIC FRACTURE MECHANICS PERSONAL COMPUTER CODE FOR NUCLEAR POWER PLANT PIPING RELIABILITY ASSESSMENT

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Design Features

Edited by D. B. Trauger

Containments for Gas-Cooled Power Reactors: History and Status

By P. M. Williams[a]

Abstract: *This article discusses containment design for gas-cooled reactors by surveying the international background and modern practices. Modern gas-cooled reactors are differentiated from earlier designs by their fuel forms, which are either of the ceramically coated particle type being developed by the United States, the Federal Republic of Germany, and Japan or of the stainless-steel-clad, uranium-dioxide pellet type used by the United Kingdom. Early reactor designs and experiences are described to provide support and background for subsequent discussions of modern designs, accident potentials, and radionuclide source terms. Resolution of containment issues is seen to require a careful balance between a mechanistic source-term approach and the traditional "defense-in-depth" requirements that add additional systems and barriers to account for uncertainties and unknowns. Designers and researchers need to strive within economic realities to achieve a degree of completeness and conservatism on the containment design that can be judged satisfactory by regulatory authorities and others.*

This article and its references present the history and status of containment designs for gas-cooled reactors to aid in establishing their prudent and appropriate design bases for containment designs. This aim is approached by first summarizing the considerable early international background in design, development, and operations and then describing the safety aspects of modern designs. Accident potentials and radionuclide source terms are next described, followed by a concluding discussion of containment selection and adequacy. Although this study considers the various types of gas-cooled reactors and the international differences and approaches, its formulation is based mainly on information and deliberations developed during the course of the Nuclear Regulatory Commission's (NRC's) recent and continuing review of the Modular High-Temperature Gas-Cooled Reactor (MHTGR),[1] proposed by the U.S. Department of Energy (DOE).[2] Note that NRC has not yet completed its review, and no final conclusions or findings are to be implied by this article.

Containment design bases are discussed herein (1) in terms of the multiple barrier concept to fission-product transport usually associated with the principle of "defense-in-depth" and (2) from a mechanistic safety analysis approach, wherein a radionuclide source term is derived from the response of the reactor system and the fuel to a spectrum of postulated accidents. In a manner similar to light-water-reactor (LWR) safety assessments, information is evaluated for both of these approaches; it will be seen that these approaches are not mutually exclusive. In comparison to LWRs, gas-cooled reactors have significant materials and transient differences to be seen to result in less familiar source-term characteristics and containment design bases. Also, and unlike LWRs, no general agreement or guide currently exists to the radioactive source term for containment design such as given in TID-14844 (Ref. 3).

EARLY DESIGN, DEVELOPMENT, AND OPERATIONAL EXPERIENCE

Early development activities for gas-cooled reactors explored many missions (research reactors; plutonium production; maritime, aircraft, and space propulsion;

[a]U.S. Department of Energy.

and electric power generation) and resulted in extensive experiences with many materials: reactor coolants of air, nitrogen, hydrogen, carbon dioxide, and helium; moderators of graphite, water, heavy water, metal hydrides, and beryllium oxide; fuel forms of uranium metal, uranium alloys, uranium oxides, dicarbides, and thoria; and claddings of aluminum, Magnox (a magnesium–aluminum alloy), stainless steel, porous and dense pyrolytic carbon, and silicon carbide. In summary, a great deal of design, materials, construction, and operating experience has existed for a long time to support the further development and licensing of gas-cooled reactors. Much of this background is described in the book *Thermal and Flow Design of Helium Cooled Reactors*.[4]

Gas-cooled reactor history effectively begins with the startup in November 1943 of the graphite-moderated, air-cooled, 3.5-MW, X-10 reactor in Oak Ridge, Tenn. The X-10 was the pilot plant for the water-cooled, plutonium production reactors at Hanford, Wash., and used open-circuit cooling (that is, the air coolant was drawn from and exhausted to the atmosphere). Commercial gas-cooled nuclear power began in 1953 when the United Kingdom decided to combine plutonium production with electric power generation, and work was started on the four-unit power station at Calder Hall. Sir Christopher Hinton, who began directing Britain's production of fissionable material in 1946, wrote that if Britain had not been a densely populated country, the water-cooled design of the U.S. Hanford reactors would have been adopted, but gas cooling was chosen because ". . . it was thought that we could do better to choose a type of pile which was inherently stable."[5] Containment for the Calder Hall reactors as well as all subsequent gas-cooled power reactors in the United Kingdom consists of two barriers: the fuel cladding and the primary system pressure boundary. These first power reactors are graphite moderated, have Magnox cladding and natural-uranium metal rods, and are cooled by forced circulation of carbon dioxide at a pressure of 100 psig and at an outlet temperature of 635°F. The Calder Hall reactors became operational in 1956, continue to produce a net electrical power of 50 MW each, and are expected to operate for a full 40-year life with a possible extension to 50 years. The U.K.'s extensive commitment to gas-cooled reactor technology has included construction of 26 Magnox reactors (22 remain in operation) and 14 Advanced Gas-Cooled Reactors (AGRs), which deliver steam at the modern conditions of 538°C (1000°F) and 16.5 MPa (2400 psi). For this reason, the U.K.'s background of operational history and component development, together with its reactor safety experience, is valued in HTGR regulatory decision making even though the respective fuel designs and gas coolant are not the same.[6] For example, the up-flow design for core cooling permits a means for passive decay-heat removal not found in the large reactor designs of the United States and Germany.

The AGRs have slightly enriched uranium oxide fuel pellets stacked and sealed in stainless steel rods. These are bundled and supported in a manner functionally similar to LWR fuel, but the AGR core power density, about 2.7 W/cm^3, is substantially less because of the graphite moderator and carbon dioxide cooling. The AGR fuel's response to overheating is relatively benign because the increased oxidation in the carbon dioxide atmosphere is expected to cause the cladding to remain in place rather than melt. This knowledge was gained from laboratory experiments, as no overheating events have been reported for the AGRs. For the most recently commissioned designs, the 625-MW(e) Heysham-2 and the 700-MW(e) Torness Point reactors, gas outlet pressures and temperatures are 4.36 MPa (632 psia) and 616°C (1140°F), respectively.[7]

France's early interest in gas-cooled reactors aided development in the United Kingdom. In 1951, the 2-MW research reactor at Saclay, which began operating with nitrogen coolant and later switched to carbon dioxide, was the first gas-cooled reactor to use closed-circuit, pressurized cooling. These experiments, coupled with the experience of the air-cooled, open-circuit, G1 plutonium production reactor, formed the basis for France's gas-cooled power reactor program. Although similar to the U.K.'s program in coolant, moderator, and fuel, the French program introduced the use of the prestressed concrete reactor vessel (PCRV),[8] which the United Kingdom adopted for its later Magnox reactors and all its AGRs. Early U.K. and French designs considered that no additional fission-product barriers were needed beyond the two barriers of the fuel cladding and the primary system pressure boundary. The United Kingdom later reaffirmed its two-barrier containment decision in developing the modern AGRs, partially based on the use of the PCRV to house its integral concept for the primary coolant system, a topic that will be further discussed.

Major accidents occurred with the early gas-cooled reactors, and lessons were learned. In England in October 1957 one of the open-circuit, air-cooled, plutonium production reactors at Windscale released an uncontrolled amount of radionuclides.This event included ignition and burning of the cladding and uranium metal fuel and transport to the surrounding countryside of an estimated 74 TBq (2000 Ci) of ^{131}I, 22 TBq (600 Ci) of ^{137}Cs, 3 TBq (80 Ci) of ^{88}Sr, and 330 GBq (9 Ci) of ^{90}Sr (Ref. 9). Almost exactly a year earlier, the Windscale accident was predicted by a lesser publicized fuel failure in the French Gl reactor when, as it was reported, "... quite suddenly a bar of uranium in one of the channels took fire."[10] The Windscale accident was a principal contributor to improved public policies for reactor safety throughout the world and illustrated the dominant role of iodine as a radionuclide source term. Although sobering, Windscale did not lead to any additional containment provisions for gas-cooled power reactors in the United Kingdom and France. No major accidents reoccurred in U.K. power reactors over the years, but the French experienced fuel melting events in four of the nine power reactors by cooled carbon dioxide that they eventually built, two of which required extensive repairs.[11] This may have been a contributing reason for France's decision to adopt LWR technology in the mid-1970s.

Although several gas-cooled reactors were built and operated in the United States during the 1950s and 1960s mainly for propulsion purposes, the helium-cooled, 40-MW(e) Peach Bottom 1 HTGR provided the major background for the continued commercialization of HTGR technology in the United States. Operated by the Philadelphia Electric Company from 1967 to 1974, this HTGR introduced coated-particle fuel and initiated the development of licensing criteria for this type of gas-cooled reactor.[12] With respect to containment design bases, an important difference between Peach Bottom 1 and subsequent HTGRs was that its carbon-coated, particle fuel was designed for "venting" iodine and other volatile fission products to the primary system rather than retention within the coated-fuel particles. The helium purification system and associated storage tanks were designed to accommodate 100% of the fission-product inventory, and a steel shell was used as a second fission-product barrier. Although Peach Bottom 1 had in effect a two-barrier containment, the fuel cladding was not one of the barriers. Consequently the accident source term did not directly relate to the fuel, and Peach Bottom 1's contribution to the containment philosophy for modern HTGRs is not direct. However, Peach Bottom 1 has supplied important information toward developing modern, coated-particle fuel and to knowledge of fission-product transport phenomena in the primary circuit.

Over the years, much international cooperation has aided in the development of coated-particle fuel, primarily initiated by the Dragon Project.[13] For modern, coated-particle fuel, development objectives are to resist high-temperature coating failure by internal fission-product gas pressure and chemical interactions, to protect from damage to the coatings by manufacturing processes when the particles are compacted into a graphite matrix fuel, and to reduce contamination by free uranium compounds in regions outside the particle coatings. Priority topics of current research are (1) to validate by statistically significant testing that the proposed product integrity specifications (e.g., thickness and density of coating layers) are sufficient to meet design requirements, (2) to quantify time-at-temperature resistance to fission-product release by the hydrolysis chemical reaction from those kernels exposed by defective coatings to moist reactor coolant, and (3) to develop a manufacturing quality control program to ensure reliability with respect to the product specifications. The coated-particle fuel is the basis for HTGR designs in the United States, the Federal Republic of Germany (FRG), and Japan. The version of particle fuel selected by DOE for the MHTGR is 0.8 mm in diameter and consists of a kernel composed of a mixture of uranium dioxide and uranium dicarbide successively coated by layers of porous carbon, dense pyrolitically applied carbon, silicon carbide, and an outer layer of pyrolytic carbon. Recently the coatings have been augmented by a seal coating of silicon carbide, a protective layer of pyrolytic carbon, and a second seal coating. These added layers further protect the coated particles when formed into a fuel matrix compact.

In the FRG, the 15-MW(e), Arbeitsgemeinshaft Versuchsreaktor (AVR) was a "pebble-bed" type of HTGR operated between 1967 and 1989 and, like Peach Bottom 1, had an initial core loading that encouraged diffusion of fission products to the primary coolant. Similarly, a steel containment vessel was provided. The AVR used particle-fueled, graphite spheres 6 cm in diameter that traveled downward through the core. The fuel spheres were later modified to contain modern, coated-fuel particles. The AVR was the main fuel development tool for the pebble-bed concept, and it and supplementary laboratory fuel testing became the major support of the FRG's position that an LWR-type containment barrier was not needed for future HTGRs (Ref. 14).

The first nuclear power reactor in Japan, which started commercial operation in July 1966, was the carbon dioxide-cooled, 166-MW(e) Tokai station located 80 miles northeast of Tokyo. The plant design generally followed the design of the U.K.'s Magnox reactors; however, because of population concerns, its containment design provided a partial third barrier for a postulated release of coolant by sealing after pressure decay "... the gas release holes and other gaps in the concrete biological shield that surrounds the reactor vessel and some of the primary system piping."[15] In 1969, the Japan Atomic Energy Research Institute (JAERI) initiated studies on the Very High-Temperature Gas-Cooled Reactor (VHTR) in recognition that a nuclear process heat source of 900°C or higher would find use in coal gasification and hydrogen and methanol production. JAERI and NRC developed an agreement in 1975 to exchange safety information,[16] and an "Implementing Arrangement" for cooperation in research was developed between DOE and JAERI in 1985. The Japanese have developed basic test facilities over the years that include the Helium Engineering Demonstration Loop (HENDEL) and a critical experiment facility. Also, they have recently begun construction of a 30-MW(t) test reactor, the High-Temperature Test Reactor (HTTR). Criticality for this reactor is scheduled for 1996, and test facilities are to be operational by 1997 (Ref. 17). Japan believes that the most promising of the HTGR applications is process heat and is making sustained progress toward this goal.[18] The containment design for the HTTR includes a steel shell as a third barrier.

In summary, early international operating experience for gas-cooled power reactors has exhibited a high level of safety. In the United Kingdom, a two-barrier containment has been judged sufficient by designers and licensing authorities in the past, and three barriers have been used in Japan. The next section discusses containments for modern designs.

SAFETY CHARACTERISTICS OF MODERN DESIGNS

The gas-cooled reactor types that this article considers "modern" are listed in Table 1, which includes reactors that have been built and operated, those planned but not built, and the HTTR test reactor in Japan. It excludes the Magnox power reactors of the United Kingdom and France, the Peach Bottom 1 reactor in the United States, and the conceptual designs in countries that have not yet actually constructed or operated a gas-cooled power reactor, such as the former U.S.S.R. The criterion for "modern" is the fuel form, which is taken as either coated-particle fuel or the U.K.'s stainless-steel-clad fuel rods.

United Kingdom

The Heysham-2 reactors will be the focus of discussion, as the United Kingdom considers these reactors and the Torness Point reactors the most promising AGR designs for future development, if such development were to occur.[19] Figure 1 illustrates the design features that determine their reactor safety characteristics. It is seen that the entire primary system is enclosed within a PCRV, which has an internal diameter of 20.4 m (67 ft). This is known as the integral primary system concept and is one of the major reasons that the PCRV was judged in the United Kingdom as a barrier sufficient to obviate the need for further containment or confinement structures.[20] The integral concept eliminates the potentials for failure to the environs of large gas ducts and thus reduces the probability and significance of depressurization or air ingress events. Further, it reduces the significance of penetration failures because of their relatively small size. A second major reason is the PCRV itself, which is judged to be advantageous compared with a steel pressure vessel because its concrete and steel tendon arrangement provides a practical means for high vessel integrity at very large volumes and a gradual and detectable failure mode. All PCRVs have a thermal barrier system against the hot reactor coolant to maintain the concrete below damaging temperatures. This generally consists of metal or fibrous insulating materials and a water-cooled metal liner against the inner concrete surface.

Coolant flow paths are indicated by the arrows in Fig. 1, where upward flow is shown through the fuel assemblies, and all other regions of the reactor are maintained at cooler temperatures by downward flow directed by the gas baffle. Hot, exiting fuel-element flow turns downward at the upper plenum to pass through the boilers and, now cooled, enters the gas circulators, where its pressure is increased. Circulator exit flow divides; some flows upward through the "diagrid" core support structure and peripheral passage to where it is turned downward by the gas baffle, and the second portion flows directly to the lower plenum where it joins the first portion before entering the fuel assemblies. The equipment is arranged in four independent loops, any one of which can remove decay heat under pressurized or depressurized conditions. In addition, an external loop can, under pressurized conditions, remove decay heat passively. With this flow arrangement, the frequency of a core heat-up event and core damage is assessed at less than 1×10^{-6} per reactor year, an estimate supported by the fact that no fuel damage by core

Table 1 Modern Gas-Cooled Power Reactors[a]

Country	Reactor name	Power level, MW(e)	Fuel form	Reactor vessel	Containment	Status
United Kingdom	14 AGRs, Heysam-2	625	UO_2 pellets SS clad	PCRV	PCRV	In operation
United States	Fort St. Vrain	330	Prismatic particle type UC_2/ThC_2	PCRV	PCRV plus confinement	Being decommissioned
	Large HTGRs	880 to 1160	Prismatic particle type UC_2/ThC_2	PCRV	Concrete dome	Reviewed by NRC, never built
	MHTGR	135	Prismatic particle type UCO	Steel	Undecided	Draft preapplication SER issued by NRC
Federal Republic of Germany	AVR	13	Pebble-bed particle type	Steel	Steel	To be decommissioned
	THTR	300	Pebble-bed particle type (U, Th) O_2	PCRV	PCRV plus confinement	To be decommissioned
	HTR-Modul	80	Pebble-bed particle type UO_2	Steel	Confinement	Design discontinued
	HTR-500	550	Pebble-bed particle type UO_2	PCRV	PCRV plus confinement	Design and licensing report prepared
Japan	HTTR	30 MW(t)	Prismatic particle type	Steel	Steel	Under construction

[a]AGR, Advanced Gas-Cooled Reactor; AVR, Arbeitsgemainshaft Versuchsreaktor, HTGR, High-Temperature Gas-Cooled Reactor, HTR, High-Temperature Reactor; HTTR, High-Temperature Test Reactor, MHTGR, Modular HTGR; PCRV, prestressed concrete reactor vessel; SER, Safety Evaluation Report; SS, stainless steel, THTR, Thorium High-Temperature Reactor.

heat-up has ever occurred in the U.K.'s gas-cooled power reactor program. This flow arrangement also has the benefit of permitting the use of metals rather than graphite for the core support structure and allowing top entry for control rods.

The maximum credible accident and the basis for reactor siting dose estimates is taken for the AGR as the dropping, fracture, and fission-product release during a refueling transfer from the reactor vessel of a full-length, fully exposed, fuel assembly. Fuel within the PCRV is considered safe under all conditions. For future AGRs, a containment structure will be provided for the refueling process but not for the reactor as a whole.

Development of the AGRs has been lengthy and difficult, and, although some of the earlier AGR types have shown poor availability, the four reactors of the type illustrated by Heysham-2 have ". . . regularly achieved load factors of 70 percent or better."[19] Many major technical problems were solved during the development history, an example of which is graphite weight and strength loss by carbon dioxide corrosion. This has been controlled by the addition of prescribed quantities of methane. Although the AGRs can be viewed as a major technical and safety achievement, the cost of the electric power they produce has been judged in England to be comparatively high and has led to the decision to build the Sizewell-B pressurized-water reactor.

United States

Three modern HTGRs are listed in Table 1 for the United States: Fort St. Vrain, the "Large" HTGR, and the MHTGR. After a lengthy startup period beginning in 1974, Fort St. Vrain began commercial operation in January 1979, but it was decided to shut down and decommission the plant in August 1989 because of several design and operational difficulties. Fort St. Vrain demonstrated excellent performance by the coated-particle fuel and provided a learning experience vital for further development of HTGR technology.[21] It followed the AGR design in the use of the PCRV and the integral cooling system but used downward flow. A containment design was accepted for licensing that depended largely on the PCRV's integrity, but a ventable confinement building with a filter system was added. This design addressed two postulated design-basis accidents. For the first, a core heat-up caused by loss of forced helium circulation, the confinement-filter system allowed an intentional depressurization maneuver that

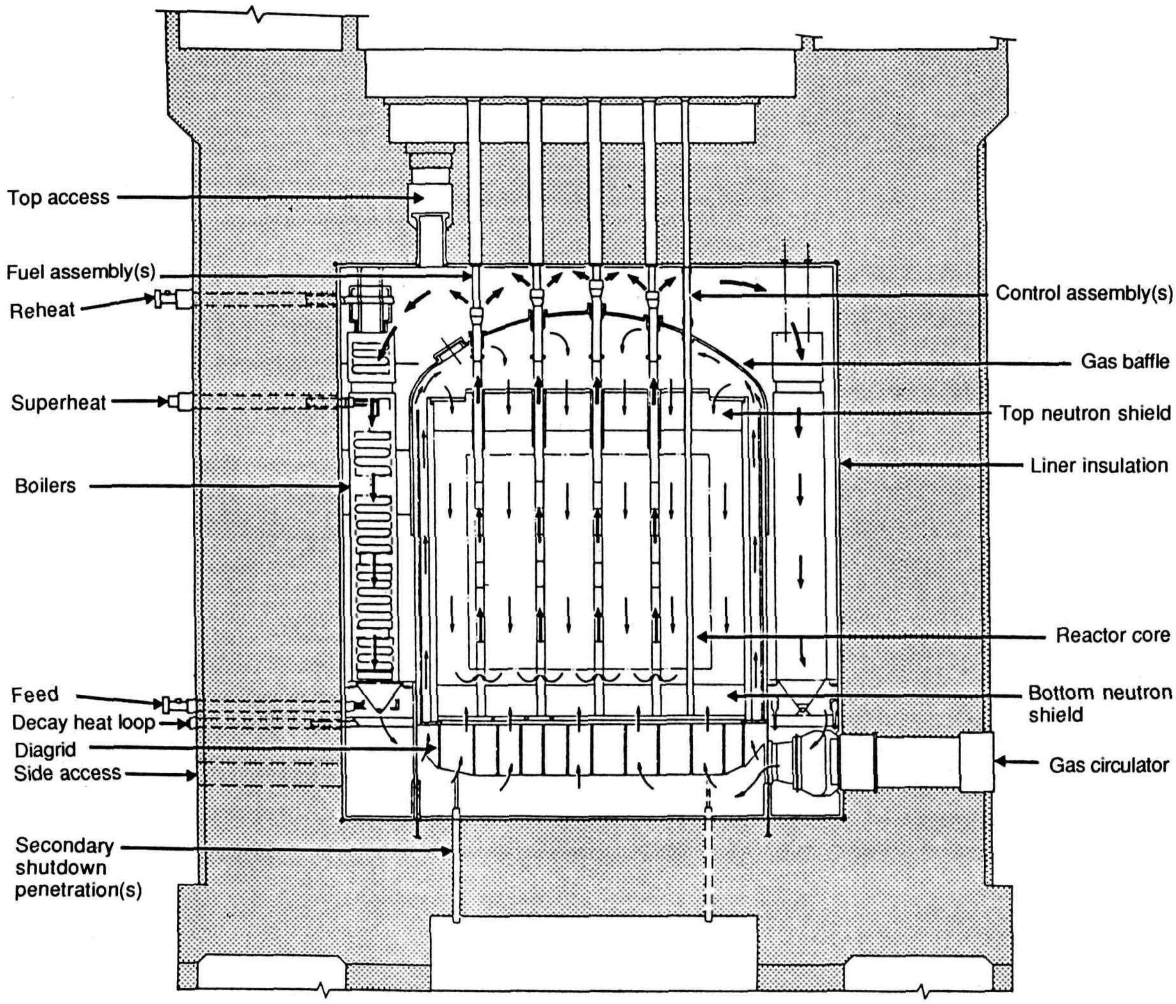

Fig. 1 United Kingdom: Advanced Gas-Cooled Reactor (AGR).[7]

would permit subsequent isolation of the PCRV to form a low-pressure containment boundary, with heat removal by the liner cooling system. The source term to be contained by the PCRV was taken nonmechanistically as that given in TID-14844 for LWRs. For the second, the rapid depressurization of the primary system, louvers of the confinement building would blow open and preserve the building integrity and the filter system. The off-site dose was computed from the unattenuated transport to the environs of a source term consisting of (1) those radionuclides circulating with the helium and (2) a lift-off fraction of 5% of the controlled inventory of plated-out radionuclides in the primary system. Core heat-up sufficient to cause fuel particle failure in the depressurization accident was designed to be precluded by continued operation of the helium circulators, driven by safety-grade auxiliary power. The source term was evaluated to be sufficiently small to meet acceptable dose criteria at the site boundary.[22]

Following the issuance of the Fort St. Vrain construction permit, design and licensing activities were performed for larger HTGRs ranging in power from 766 to 1160 MW(e), for a 300-MW(e) fast breeder reactor, and for a direct-cycle, helium turbine plant.[23–27] With the exception of the fast gas breeder reactor, which used rod-type fuel similar to liquid-metal reactor fuel, the reactor designs were generally similar to Fort St. Vrain except for the steam generator and circulator arrangements and the use of auxiliary circulators and heat exchangers for decay-heat removal. With respect to the containment design, however, all these concepts were provided with third barriers in the form of large, steel-lined concrete domes similar to those for LWRs and used a delayed version of the TID-14844 source term. A study was made of available data supporting a mechanistic source term,[28] but it was concluded that a larger data base was needed. None of these designs was ever built, and the last major study for this concept was completed in 1984 (Ref. 29).

The MHTGR is shown in Fig. 2. The cooling circuit of helium, graphite moderation, and the coated-particle fuel form is preserved, and like its predecessors, helium flow is downward past the fuel. The principle difference is that a passive means for decay-heat removal under both pressurized and depressurized conditions is provided by the use of a steel reactor vessel and a naturally convective air-flow system. The reactor vessel, 6.83 m (22.4 ft) in diameter, and a separate steam generator vessel are located in an underground cavity or silo connected by a concentric-flow cross-duct vessel. An air-cooled natural convective system in the reactor vessel cavity provides a sink to remove heat transmitted from the surface of the uninsulated reactor vessel. Air flow is provided by the reactor cavity cooling system (RCCS), which has inlet and outlet duct structures above grade level and is not shown in the figure. The air flow is provided passively at all times and, in the emergency situations of loss of helium flow or pressure, the vessel temperature rises to transmit decay heat, mostly by radiation, to the RCCS. The required amount of energy to be transmitted is that energy sufficient to ensure that the fuel-particle temperature will not exceed 1600°C and that the reactor vessel will not exceed the allowable limits of temperature and time (1000°C and 1000 h). Because the commercial MHTGR could potentially remove decay heat passively without causing appreciable failure of the fuel particles to retain radionuclides, the designers have proposed that it should not require a third containment barrier. For the desired performance, a balance must be struck between the heat transport conditions within the reactor vessel and core, the allowable temperature of the steel vessel, and the amount of vessel surface area to achieve the necessary amount of heat transport. The balance results in a limitation of the surface-to-volume ratio of the reactor and accounts for the long annular configuration of the core.

A consequence of this design is that the overall electric power that can be delivered from a single reactor is limited to the range 135 to 175 MW(e). Electric power stations are expected to use these reactors in multiples of factory-built, standardized units, and hence the concept is called "modular." Although this design offers important advantages in passive safety, its economic competitiveness is of concern and is under study.

Federal Republic of Germany

On the basis of early experience with the AVR pebble bed reactor and its continued availability for fuel and other development projects, the FRG designed, constructed, and operated the 300-MW(e) thorium *high-temperature reactor* (THTR), *the principal features* of which are shown in Fig. 3 (Ref. 30). The THTR used the integral cooling circuit within a PCRV, and, like the U.S. HTGRs, flow was downward with the core supported by structural graphite from below. Fuel spheres were fed at the top and removed below while the reactor operated. The system for emergency removal of decay heat contained, among its provisions, two independent cooling loops powered by dedicated diesel generators. The principal source term for reactor siting was taken as the unmitigated rupture of the largest tube carrying reactor helium (65-mm diameter) at a pressure of 40 bar and a temperature of 260°C. A rectangular reactor building capable of 1.6-bar overpressure provided a third barrier confinement function for major helium leaks by controlled discharge to the ventilation stack, with smaller releases being filtered first. The emergency decay-heat removal system was designed to prevent radionuclide release by fuel damage.

After a testing and startup period that verified design calculations, the THTR operated commercially from June 1987 to October 1989, when it was shut down for inspection. A small fraction of the bolts holding thermal barrier cover plates in the hot gas ducts had failed, and, although resolution of this safety issue had been achieved, both the state and federal governments decided to no longer support repair and operating costs. These decisions resulted in plans to decommission the reactor.[6] In addition to the AVR and THTR projects, FRG gas-cooled reactor design and research activities have resulted in a design for the HTR-500, a larger and more economical version of the THTR, and a design for the HTR-Modul, a design similar to the MHTGR but of lower power.[31,32] Commercialization of these designs is not proceeding at present for economic reasons, although the safety review for the HTR-Modul by licensing authorities was proceeding well.

Japan

The HTTR is illustrated in Fig. 4, where it is shown that the graphite core is housed in a steel reactor vessel with heat exchangers, other equipment, and connecting piping, all within a steel containment vessel. The surrounding building is maintained at slightly negative

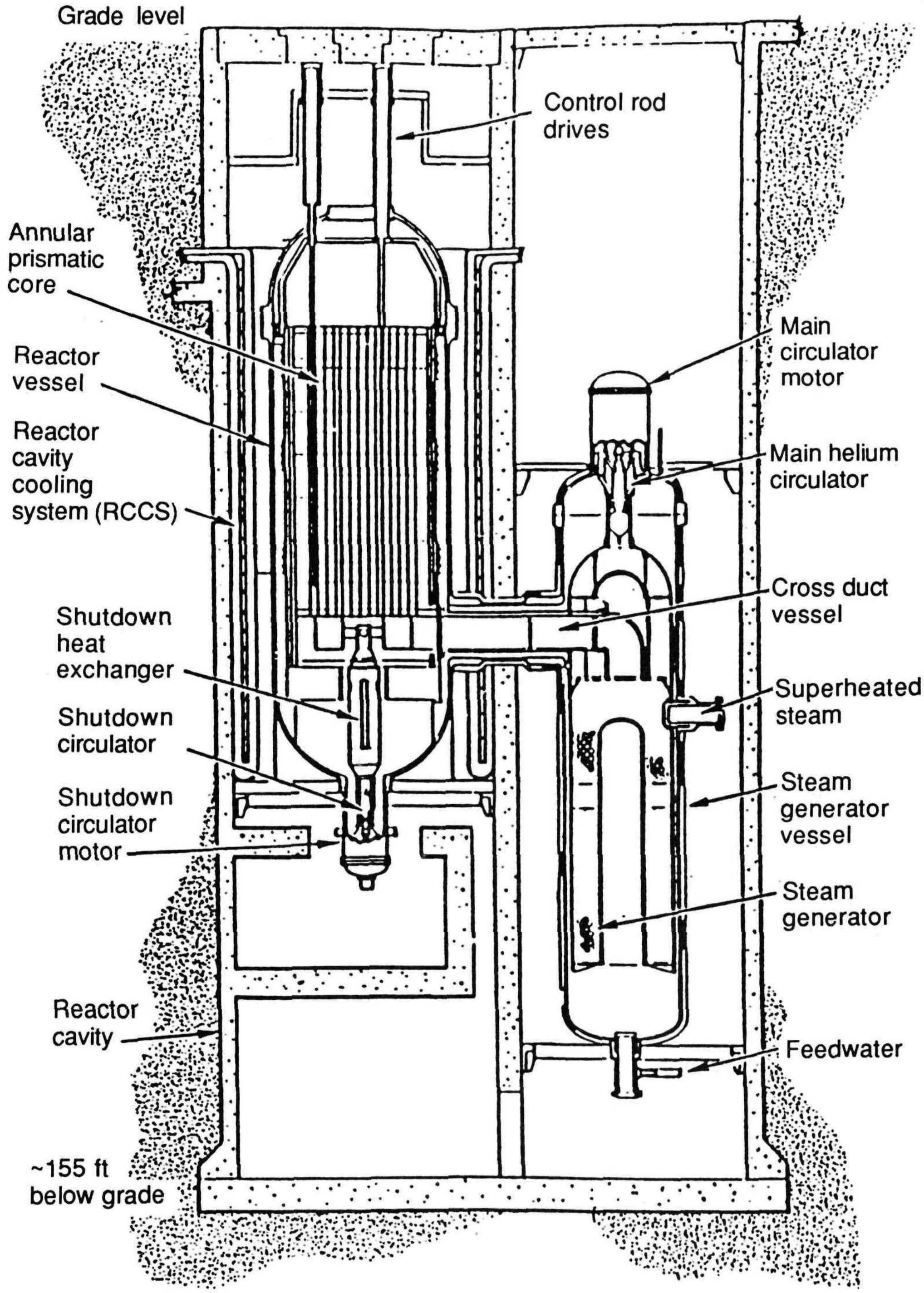

Fig. 2 United States: Modular High-Temperature Gas-Cooled Reactor (MHTGR).[18]

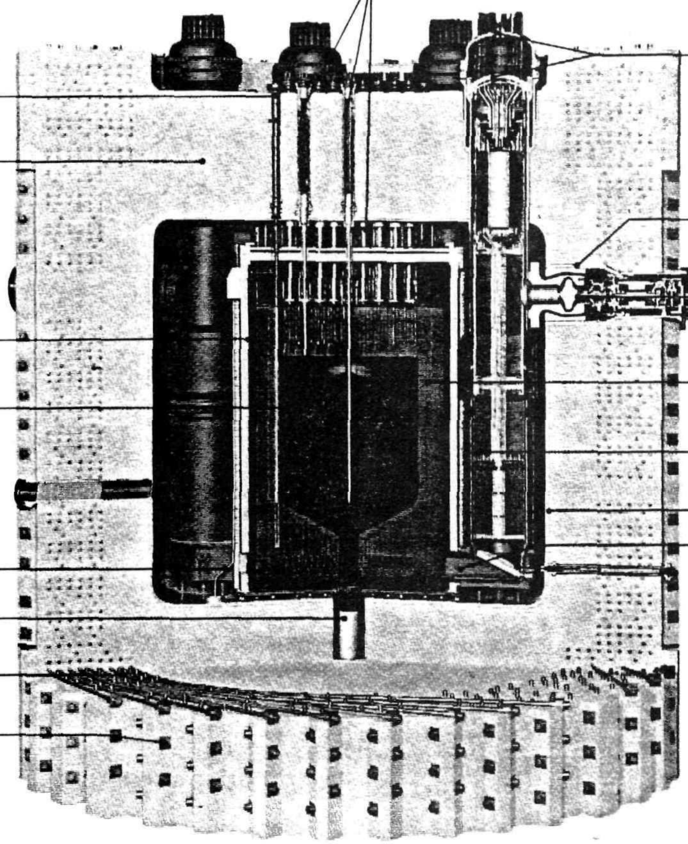

Fig. 3 Federal Republic of Germany: Thorium High-Temperature Reactor (THTR).[30]

pressure to the environment by ventilation systems and serves both as a "service area" and a confinement building. The containment vessel design functions contain fission products and limit the amount of air that could "possibly react with graphite in the reactor core on an accident."[33] The off-site radiation dose estimate is based on a primary system pipe rupture and is stated to be "remarkably reduced by the containment vessel together with the confinement."

In summary, development of modern gas-cooled reactors remains active, but construction activities in the foreseeable future are uncertain except for Japan. In the United States, a version of the MHTGR for the production of weapons-grade tritium is being studied, with a decision to proceed scheduled for August 1993. All research and design needs common to the military and commercial versions of the MHTGR are expected to be performed by the defense project and kept unclassified.

ACCIDENT POTENTIALS

Table 2 is a matrix developed from many of the references cited; it lists five major potential accidents with a summary of concerns and mitigation approaches considered by reactor designers and regulators in the United Kingdom, the United States, the FRG, and Japan. The accidents discussed are selected to illustrate the containment needs of gas-cooled reactors but are not inclusive of all low-probability events that could be postulated. The accidents selected include those commonly found to result in the maximum for off-site dose estimates, those judged to be inclusive of lesser accidents, and those which illustrate differing treatments of safety issues. Not listed are accidents to be precluded by structural design, inspection, and maintenance (e.g., control-rod ejection, catastrophic reactor vessel failure, and core support collapse from a large seismic event).

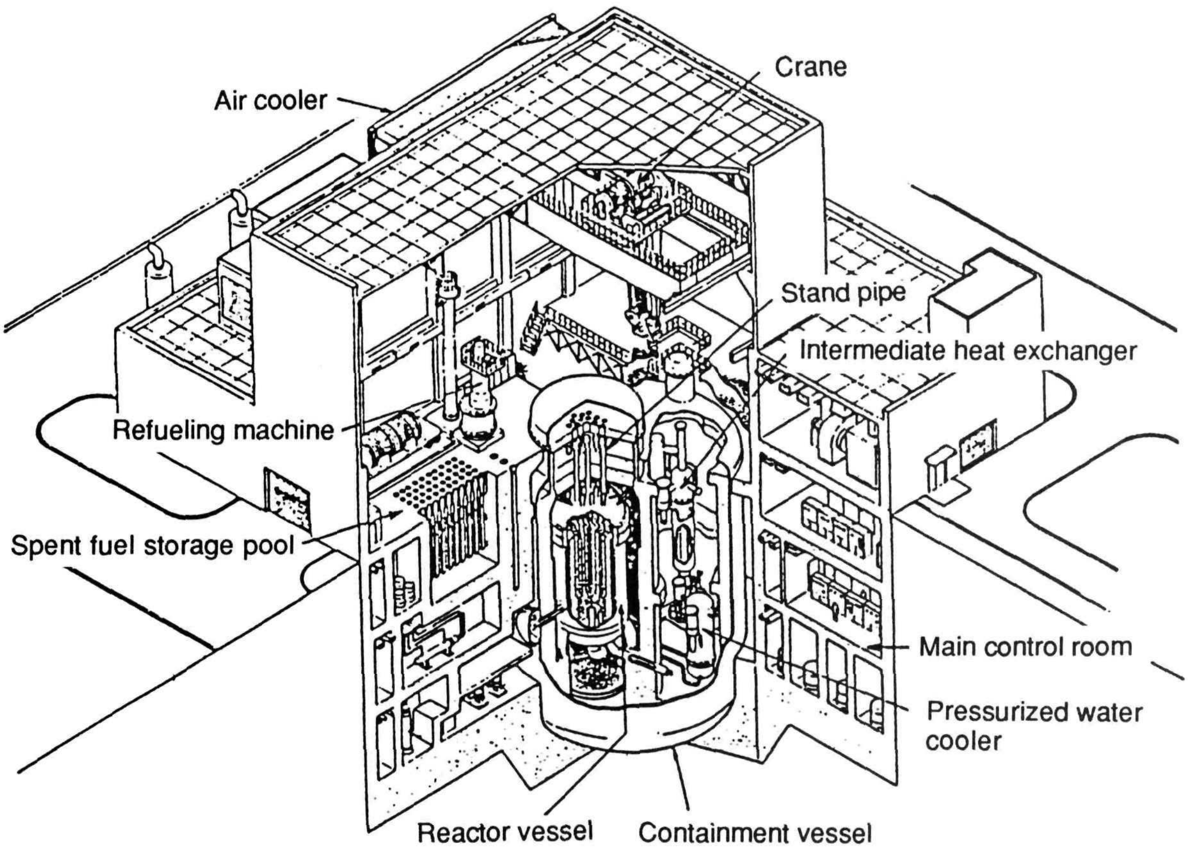

Fig. 4 Japan: High-Temperature Test Reactor (HTTR).[18]

Also not listed are examples of initiators of accident sequences as could be caused by anticipated operational occurrences or unanticipated gross human error. Although the potential and consequences for such events need to be carefully examined by both designers and regulators, the transient behavior of gas-cooled reactors is comparatively sluggish because of their low power densities and high heat capacities. Because this characteristic enhances recovery opportunities, the consequences of lesser initiators and human errors are, for purposes of source-term identification and containment design, judged bounded by the major accidents. This forgiving response has been examined with favorable results by various probabilistic risk assessments and supported by operating experience.

The accidents selected in Table 2 are not divided into design basis and severe accidents as would be customary in a discussion for LWRs. A severe accident for a gas-cooled reactor would be sustained heating or chemical attack of the fuel followed by release of radionuclides that would be greater than could be accommodated by operating, safety, mitigative, and siting features of the design. For the MHTGR, the design requirement is to preclude such accidents by a probability of less than 5×10^{-7}. This is to be achieved by reactor safety through design integrity and reliability requirements on the fuel rather than by reliance on a traditional, high-pressure, low-leakage containment guidance.

Steam Generator Failure

Steam generator failures have occurred and should be conservatively considered in the safety analysis. Although (1) gas-cooled reactors are typically provided with highly reliable moisture detection systems and fast-acting valves to isolate and dump a failed steam generator and (2) the most current designs allow for

Table 2 Concern and Mitigation Summary[a]

Selected accidents	United Kingdom (AGR)	United States (FSV and MHTGR)	Federal Republic of Germany (THTR and HTR-Modul)	Japan (HTTR)
Steam generator failure	Causes reactor pressure increases, mitigation by isolation and relief valve action, failed-open relief valve can depressurize system	Same mitigation means as for AGR, but reactivity and fuel damage major concern in both U S and FRG for modular designs Experiments needed for resolution		Same mitigation means as AGR, steel containment mitigates fuel damage concern
Rapid depressurization of reactor coolant	Not of direct concern, as no appreciable inventory of circulating or plated-out fission products exists	Lift-off and release of plated-out fission products in primary system		
		FSV Unmitigated release MHTGR Unmitigated release (under study)	THTR and HTR-Modul Release mitigated through system to building stack	Release mitigated by containment and confinement
Loss of forced circulation	Pony motors or natural convection prevent core heat-up	FSV Emergency depressurization followed by core heat-up in subsequently sealed PCRV MHTGR Passive heat-removal system limits core temperatures to below fuel failure threshold	THTR Not considered credible, backup cooling machinery provided HTR-Modul Passive heat-removal system limits core temperatures to below fuel failure threshold	Not considered credible, backup cooling machinery provided
Graphite fire	Not considered credible	Considered highly unlikely, but concern not fully resolved	Silicon carbide coating on fuel exterior being considered	Precluded by steel containment building
Transfer of irradiated fuel	Drop, fracture and melting of irradiated fuel assembly outside PCRV Future AGRs would provide a structure to contain releases			Coated-particle fuel form reduces fuel handling concerns

[a]AGR, Advanced Gas-Cooled Reactor, FSV, Fort St Vrain, HTR, High-Temperature Reactor, HTTR, High-Temperature Test Reactor, MHTGR, Modular High-Temperature Gas-Cooled Reactor, PCRV, prestressed concrete reactor vessel, THTR, thorium high-temperature reactor

full inspection and ready replacement of steam generators, it is international practice to recognize that steam and water ingress events cannot be precluded by design. Examples of steam ingress are pin-hole tube leaks in Fort St. Vrain and a tube rupture at a bimetallic weld in the Hartlepool AGR. A large water ingress event occurred when the overhead steam generator in the AVR drained undetected through the core to the bottom of the reactor vessel when the reactor was in a shutdown condition. Steam and water ingress events can lead to the following states in a progression toward a major accident: (1) overpressurization and subsequent depressurization of the reactor primary system by a failed-open pressure relief valve, (2) a reactivity insertion and a subsequent power excursion for the modular designs in the United States and the FRG, and (3) possible radionuclide release from defective fuel by the hydrolysis chemical reaction.

In performing the safety analysis, credit for reactor trip is usually given and is based on many available signals and, ultimately, on a signal of high pressure. For the modular designs, steam ingress could result in a power excursion followed by shutdown initiated by the Doppler reactivity effect. The nature and consequences of the power excursion should be addressed by reactor physics research.

The generation of the combustible gases of hydrogen and carbon monoxide by the reaction of hot carbon and water vapor is also a major accident progression state. The consequences of subsequent combustion in a location favorable to aggravation of a serious event already in progress need to be considered in the accident analysis and, if necessary, precluded by design.

Rapid Depressurization of Reactor Coolant

Rapid depressurization is a design-basis accident for gas-cooled reactors because its consequences entail (1) deficiencies in heat removal that can threaten fuel integrity, (2) transport beyond the primary system pressure boundary of the inventory of fission products circulating with the reactor coolant or plated out on various surfaces within the primary coolant system, and (3) possible structural damage by fluid body forces to safety-related reactor internals. No rapid, or even slow, unintentional depressurization of any gas-cooled power reactor has been recorded, and on the basis of current design, inspection, and maintenance capabilities, rapid depressurization is judged a very low-probability event. Nevertheless, it is still a credible event, and its occurrence should be considered conservatively in the safety analysis.

Loss of Forced Circulation

Loss of forced circulation can occur as a result of a mechanical or power supply failure to the normally operating gas circulators, and, if all circulators become inoperative, an alternate means is needed for decay-heat removal to prevent fuel damage. For the AGRs, emergency decay-heat removal can be accomplished by "pony" motors on the existing circulators or passively, if the system remains pressurized, by natural convection. In the modular designs, it can be removed by an auxiliary system designed for that purpose or passively under either pressurized or depressurized conditions as previously described. For the large U.S. HTGR designs and the FRG THTR, fully independent auxiliary circulators and heat transport systems have been provided because passive heat removal appears not to be practical at these sizes and with downward flow. Temporary losses of all forced circulation have been experienced by Fort St. Vrain, but because of the short durations (up to about a half hour) and the very slow rise in temperature of the uncooled core, fuel damage was never of concern. However, an evaluation was made of the effect of the elevated helium temperature on metal components to establish that there was no potential for damage. Although no instances have occurred of sustained loss of forced cooling for gas-cooled power reactors, this low-probability event should be considered credible when not precluded by design provisions because of its severe consequences.

Graphite Fire

Graphite fires occurred at Windscale, as previously mentioned, and at Chernobyl, but in both cases they were ignited by combustible metals and not believed to have been self-sustaining.[34] The U.S. design has been analyzed, and graphite fires appear to be precluded on the basis of integrity requirements of the reactor vessel and the large flow resistance presented by the long core-cooling channels. Design precautions are being considered in the FRG, where the outer surface of the spherical fuel elements would be coated with a layer of silicon carbide. In Japan, it was previously noted that the design basis for the HTTR containment structure is stated to include graphite fires. Graphite fires are not considered as a necessary design basis in the United Kingdom. Analysis of natural convective studies in Japan and the conduction of chimney-type experiments in the United States may be necessary to finally resolve this issue, however.

Transfer of Irradiated Fuel

For the AGRs, the transfer of an irradiated fuel element, a portion of which occurs outside the containment barrier of the PCRV, has been found to be a potential accident with consequences controlling off-site dose estimates. As noted previously, the risk and consequences would be mitigated by a containment structure enclosing the fuel transfer area if future AGRs were to be built.

RADIONUCLIDE SOURCE TERMS

The review of accident potentials reveals some apparent inconsistencies in the selection of accidents and the treatment of their consequences as bases for containment design and for off-site dose estimates. This is partly due to differences in plant features, but the nuclear safety and economic objectives of the individual nations also appear to play an important role. The U.K. gas-cooled reactor experience indicates that it is possible to design and license a two-barrier containment system provided that the occurrence of significant fuel failure within the reactor vessel can be prevented by design and that vessel failure itself is sufficiently remote. For example, although the large steam or water ingress event is a resolved safety issue for the AGRs, this issue for the U.S. and FRG modular designs is of major consideration in establishing the containment design, and resolution is expected to require major research programs.

In the United States, the selection and uses of radionuclide source terms for advanced reactors take general guidance from the NRC's Advanced Reactor Policy Statement,[35] which provides for early interaction between the designer and regulator, with the objective of achieving licensing stability. Enhanced reactor safety is expected by such design attributes as passive means for shutdown and decay-heat removal, longer time constants, simplified safety systems, minimization of severe accidents and their consequences, reductions of exposures to plant personnel, incorporation of the defense-in-depth philosophy by maintaining multiple barriers against radionuclide release, and the use of design features either proven by existing technology or that can be established by a suitable development program. This guidance does not mandate a three-barrier containment structure, although the defense-in-depth principle is to be maintained.

A siting-basis source term for the MHTGR derived from a steam generator failure as the initiator for the MHTGR has been described by the designer[36] as comprising prompt and delayed components. This source term was mechanistically derived by combining a postulated steam ingress event that results in rapid depressurization through a failed-open relief valve combined with a core heat-up event. The prompt component originates from rapid depressurization and is proposed to consist mainly of the lift-off and wash-off of radionuclides deposited within the primary system. The normal inventory of radionuclides circulating with the reactor coolant before depressurization makes a very small contribution. The delayed component results mainly from fuel failure caused by the hydrolysis chemical reaction during the core heat-up stage, when decay heat is postulated to be passively removed. The NRC staff informed DOE "that the two source terms now used for the containment design basis . . . do not appear to adequately span the range of credible possibilities and, moreover, may not be appropriately conservative."[37] This finding is based principally on the need for further research to (1) quantify lift-off and wash-off phenomena, (2) establish the time and temperature dependence of the hydrolysis reaction with respect to the moisture concentration and temperature, (3) validate by testing the fuel design specifications, and (4) establish the practicality of a fuel quality control program that could limit the manufacture of subspecification fuel particles to the very low levels used in the safety analysis. The amount of subspecification fuel relates to the amount of radionuclides assumed to be released. The time for release needs to be known to determine whether significant hydrolysis-caused releases should be associated with the prompt as well as the delayed component of the source term.

A similar source-term concern exists for the HTR-Modul. The FRG design provides for a two-barrier, pressure-retaining containment, and, like the THTR, a reactor building that can direct radionuclide releases through filters and to a stack. The FRG designers were reluctant to provide a high-pressure containment building because such a building might unnecessarily raise both construction and operation costs to the level where the HTR-Modul would not be economically competitive. Some believe that the present design offers sufficiently improved safety and, if costs are kept modest, that the HTR-Modul has a viable potential as an attractive power source in third-world nations.

The design bases, the use of a steel containment vessel for the HTTR, and previous containment considerations for the Tokai station indicate that Japan may continue using traditional containment buildings for gas-cooled reactors. Designers, however, are expected to propose to regulators a mechanistically based containment design.[17]

CONTAINMENT SELECTION AND ADEQUACY

Currently, the containment issue for gas-cooled reactors appears well-defined in the United Kingdom, but not for the United States, the FRG, or Japan. The mechanistic source-term approach is attractive as the sole basis for containment design, particularly from economic considerations, but needs to be balanced against the designers' and researchers' abilities to achieve a degree of completeness and conservatism that can be judged satisfactory by regulatory authorities and others. Despite whatever reliance may eventually be placed on a nonmechanistic basis for containment design, adequate analytical and experimental investigations into potential source terms must be accomplished. By adequate, it is meant that phenomenological and analytical knowledge should be developed to the point at which the technical issues involved in decision making are soundly based. Furthermore, it is important that the incentives for the designer, researcher, and regulator to perform rigorous source-term investigations be maintained. The approach that would view a traditional containment structure (high-pressure, low-leakage capabilities) as capable of compensating for areas of phenomenological ignorance should be avoided.

In addition to source-term phenomena, economic tradeoffs may influence containment design. An example for the MHTGR is if the containment function to be provided by very high-quality fuel particles, compacts, and elements might be provided less expensively by a conventional containment building and fuel assemblies of lesser quality. Ideally, both the very high-quality fuel and the traditional containment building could be provided, but such is not likely in the competitive energy market.

A possible compromise for the MHTGR would take advantage of the prompt and delayed characteristic of its radionuclide source term. This design would use an initially vented reactor building that could later be effectively closed, when at low pressure, to provide sufficient containment for postulated amounts of delayed fission products. This arrangement would take advantage of the low level of fission-product inventory available for prompt transport from circulating radionuclides or plated out on primary system surfaces yet provide protection from any residual risk that could be associated with unexpected fuel failure as fuel temperatures rise over time in accordance with the decay-heat removal provisions. Such a vented containment concept in combination with reliable, high-integrity fuel has the additional advantage of eliminating postulated rapid reactor depressurization, steam-line break events, and the problems of local high temperatures from helium stratification as containment design bases and focuses directly on the mitigation of transport of radionuclides to the environment.

The final resolution of the containment design questions for advanced gas-cooled reactors is not easy or evident at present. Fortunately, decisions are not needed immediately for commercial designs, and time is available to evaluate forthcoming international research results together with possible design alternatives that may yield containment designs that can well meet both safety and cost objectives. One of the most important tools in this evaluation is believed to be augmenting the international activities currently established by the gas reactor community. A summary of the main points that influence gas-cooled reactor containment design follows:

1. Significant differences in transient response and materials of construction give gas-cooled reactors the potential for alternate approaches to the key issues of selection and analysis of postulated accidents, the radionuclide source-term definition, containment design, and emergency planning in comparison with LWRs. The potential safety advantage of advanced gas-cooled reactors is their comparatively forgiving response to transients and accident potentials. These derive from (1) their slow response to core heat-up events because of low core-power densities, (2) the very high temperature the fuel can withstand before fission-product release, and (3) the chemical inertness of the helium coolant. However, and as described within this article, challenges to these advantages exist that should be considered in safety analyses.

2. Steam ingress for the MHTGR appears to be a priority issue for resolution by analysis and experiment before a suitable containment design can be established. This issue pertains to the steam generator failure potential, the effectiveness of instrumentation and equipment designed to prevent a major steam ingress, a possible significant positive reactivity effect, and the potential for augmentation of early fission-product release from the fuel by hydrolysis.

3. The extensive research and engineering base, design and construction experience, and operational history for the AGRs in the United Kingdom provide invaluable support for the development in the United States and in other nations of HTGR components, systems, safety criteria, and safety analyses, even with the notable differences in coolant and fuel. In particular, and with regard to key safety issues, the United Kingdom provides the most extensive data base by far for the selection of accident initiators and affords data relevant to the probabilistic risk assessment of the failure potentials of steam generators, circulators, instrumentation, and other safety-related equipment. Further, the AGR design illustrates the attractiveness of such design alternatives as upward core flow and the uses of the PCRV as a containment vessel.

4. Graphite fires are not likely to be credible for modern gas-cooled reactors, but further research is desirable, mainly from a public assurance standpoint, to demonstrate this and to show that graphite fires need not be part of the containment design basis.

5. Development of HTGRs, to date, has involved a high degree of international cooperation, although the

United States, the FRG, and Japan have separate designs and objectives In the interests of minimizing development costs and achieving a higher degree of public acceptance, it may be desirable to formulate an international project that focuses on a single concept Even if a single focus cannot be achieved, international cooperation should be continued and strengthened, where possible, and particularly in the area of safety

REFERENCES

1 P M Williams, T L King, and J N Wilson, *Draft Preapplication Safety Evaluation Report for the High-Temperature Gas Cooled Reactor*, Report NUREG 1338, 1989

2 F A Silady and A C Millunzi, Safety Aspects of the Modular High-Temperature Gas Cooled Reactor, *Nucl Saf*, 31(2) 215-225 (April–June 1990)

3 U S Atomic Energy Commission, *Calculation of Distance Factors for Power and Test Reactor Sites*, Report TID 14844, 1962

4 G Melese and R Katz, *Thermal and Flow Design of Helium Cooled Reactors*, American Nuclear Society, La Grange Park Ill, 1984

5 Sir C Hinton, The Graphite Moderated, Gas-Cooled Pile and Its Place in Power Production, in *Proceedings of the International Conference on the Peaceful Uses of Atomic Energy* Vol 3 p 322, Geneva, 1955

6 P M Williams, *Report of Foreign Travel to the U K France and the FRG Pertaining to Gas Cooled Reactor Safety and Technology*, U S Nuclear Regulatory Commission, 1990

7 G C Dale and J Bowerman (Eds) *The Safety of the AGR*, Central Electricity Generating Board, London, South of Scotland, Electricity Board, Glasgow, 1982

8 M Pascal et al , General Characteristics and Original Aspects of Reactors G2 and G3, in *Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy*, Vol 8, Part 1, p 329 Geneva, 1958

9 T J Thompson and J G Beckerley, *The Technology of Nuclear Reactor Safety*, The MIT Press, Cambridge, 1973

10 M de Rouville, M Pascal, and M Scalliet, Experience Obtained During Two Years' Operation of Reactor Gl, in *Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy*, Vol 8, Part 1, p 18, Geneva, 1958

11 D Bastien, Twenty-Nine Years of French Experience in Operating Gas-Cooled Reactors, in *Design Requirements Operation and Maintenance of Gas-Cooled Reactors Proceedings of a Technical Committee Meeting*, Report 1WGGCR-19 (CONF 8809150-), pp 113-119, 1989

12 R A Clark and P M Williams, Licensing Criteria for HTGRs in the United States, in *Thermal Reactor Safety* Vol 1, Report CONF-770708-, pp 302-320, 1977

13 F P Ashworth et al , *A Summary and Evaluation of the Achievements of the Dragon Project and Its Contributions to the Development of the HTR*, Dragon Project Report 1000, U K Atomic Energy Establishment, Winfrith, November 1978

14 W Kroger, H Nickel, and R Schulten, Safety Characteristics of Modern High-Temperature Reactors Focus on German Designs, *Nucl Saf*, 29(1) 36 48 (January–March 1988)

15 I Ipponmatsu, T Yoshioka, and S Tsujimoto, Construction Experience on Tokai Atomic Power Station, in *Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy*, Vol 5 p 28, Geneva, 1965

16 Y Kurihara, Welcome Address, in *Proceedings of the Second U S –Japan Seminar on HTGR Safety Technology*, Vol 1, Fuji, Japan, 1978

17 S Shiozawa, Public Response to Questions, in *IAEA Specialists Meeting on the Behavior of Gas-Cooled Reactor Fuel Under Accident Conditions* Oak Ridge, Tennessee, November 1990

18 Gas-Cooled Reactor Associates, *Summary of Gas Cooled Programs*, GCRA Report 90 002, San Diego, January 1990

19 J Wilson, Advanced Gas-Cooled Reactor Research and Development, *ATOM*, 386 (December 1988)

20 G R Bainbridge, L F Bowles, L Cove, J R Harrison, and J M Yellowlees, The Safety of UK Gas-Cooled Reactors—A Review, in *Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy* Vol 13, p 341 Geneva, 1965

21 C F Bonilla, preface, The Fort St Vrain High Temperature Gas-Cooled Reactor, *Nucl Eng Des* 72(2) 1982

22 U S Atomic Energy Commission, *Safety Evaluation by the Division of Reactor Licensing in the Matter of Public Service of Colorado Fort St Vrain Nuclear Generating Station*, Docket No 50 267, 1972

23 U S Atomic Energy Commission, *Committee on Reactor Safeguards Gulf General Atomic Conceptual Design and Design Bases for a 1100 MWE HTGR*, Division of Reactor Licensing, 1969

24 U S Nuclear Regulatory Commission, *Safety Evaluation of the Summit Power Station Units 1 and 2 Docket Nos 50-450 50 451*, Report NUREG 75/004, 1975

25 U S Nuclear Regulatory Commission, *Safety Evaluation of the Fulton Generating Station Units 1 and 2 Docket Nos 50-463, 50 464* Report NUREG-75/015, 1975

26 U S Atomic Energy Commission, *Evaluation of the Gas Cooled Fast Breeder Reactor General Atomic Company Project No 456*, Directorate of Licensing, 1974

27 J C Scarborough, *Technology Assessment of the Gas Turbine–High Temperature Gas Cooled (Helium) Reactor*, Report EPRI-NP-805, 1978

28 M Tokar, *Evaluation of High Temperature Gas Cooled Reactor Fuel Particle Coating Failure Models and Data* Report PB-259976 (NUREG-0111), 1976

29 H J Reilly et al *Preliminary Evaluation of HTGR Severe Accident Source Terms*, Report EGG-REP 6565, 1984

30 Hochtemperatur-Kernkraftwerk GmgH, *THTR 300 Nuclear Power Plant Hamm Uentrop*, Publication No HKG 001 89 E, Hamm, FRG, 1989

31 W Steinwarz, N Kirch, and E Rohler (compilers), The Status of German High-Temperature Reactor Development, *Nucl Eng Des* 121(2) (1984)

32 W Froling and G Ballensienfen (compilers), Special Issue on the High-Temperature Reactor and Nuclear Process Heat 1984 Applications, *Nucl Eng Des*, 78(2) (1984)

33 S Saito and Y Miyamoto, Testing and Safety Considerations in the Japense High-Temperature Engineering Test Reactor (HTTR) Paper Presented at Winter Meeting, American Nuclear Society, San Francisco, 1989

34 U S Nuclear Regulatory Commission, *Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States*, Report NUREG 1251-Vol 1, 1989

35 P M Williams and T L King, *Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants*, Report NUREG-1226, 1988

36 S B Inamati, A J Neylan, and F A Silady, *MHTGR Radionuclide Source Terms for Use in Siting*, Report GA-A-19674 (CONF-8906184-1), San Diego, 1989

37 Letter to S Rosen, Director, Advanced Reactor Programs, DOE, from B M Morris, Division of Regulatory Applications, NRC, May 9, 1990

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Environmental Effects

Edited by B. A. Berven

Indoor Radon: A Natural Risk

By N. H. Harley[a] and J. H. Harley[b]

Abstract: *Radon decay products have produced excess lung cancer mortality in several groups of underground miners. Radon is the common etiological factor in several types of mines, and the data show a distinct exposure-effect relationship. Indoor radon exposures in some homes and other buildings may exceed those allowed for miners, and remediation is desirable in those cases. Extrapolation of miner risk factors predicts that exposures at average indoor concentrations will produce several thousand lung cancer deaths per year. Unlike other radioactivity contamination problems, decisions and costs for remediation are the responsibility of the individual homeowner.*

The study of natural radioactivity has always been highly interesting with its complex decay chains and equally complex chemistry of the elements involved. In addition, natural activity has been a benchmark for comparison with various man-made radionuclides in man and the environment. This has extended to natural radiation levels being offered as justification for environmental releases. For example, both authors have spent considerable time estimating the radium, thorium, and uranium content of soils around Rocky Flats when government agencies and elected officials wished to minimize the political impact of plutonium contamination. A relevant quote is from the 1990 Recommendations of the International Commission on Radiological Protection: "The fact that a man-made practice involving radiation causes doses which are small in comparison with the background does not necessarily imply that the practice is justified, but it does imply that the radiation risk situation of the exposed individual is not significantly changed by the new practice."[1]

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Since about 1980, however, studies of the direct risks of exposure to natural radiation have been dominant. Earlier work on high gamma background areas in India and Brazil showed no health effects from this exposure, probably because of small populations, short life spans, and poor epidemiological data. In these cases the exposures were two to three times the world average. When indoor radon studies began, it became apparent that a number of people were exposed to radon concentrations 10 or 100 times above the average. In fact, some indoor concentrations were higher than those allowed in uranium mines.

The flurry of interest led to the development of a considerable radon industry offering measurements and remediation and to a government effort supporting radon research that produces extensive literature.

This review will summarize the background information on radon exposures and risks, specifically for ^{222}Rn, and will indicate why there is considerable scientific controversy on the possible effects of exposure in the home. As is customary, the term radon will be used to include exposure to both the gas and its short-lived decay products.

MINER EXPERIENCE

Experience with several groups of underground miners has shown a significant excess of lung cancer among those exposed to radon, even when smoking habits are taken into consideration. The risk comes in this chain are alpha emitters and can deliver a considerable radiation dose to cells lining bronchial airways. These areas are the major site of lung cancer in the miners as well as in those exposed to other carcinogens through smoking.

The risk of death from lung disease for underground workers in some mines has been known for centuries. The high mortality among miners in the mountains bordering Czechoslovakia and Germany was noted by Paracelsus[2] in 1567 and Agricola[3] in 1597. The cause was identified as lung cancer by Harting and Hesse[4] and attributed to radon exposure by Ludewig and Lorenser.[5] Lorenz,[6] however, claimed as late as 1944 that the cause was exposure to arsenic and general poor health.

When uranium mining became important in the United States in the 1940s, some radon measurements were made, but the levels were not exciting compared with the 120 000 Bq m^{-3} (3000 pCi L^{-1}) existing in the old European mines. A health study of the Colorado miners was begun in the mid-1950s, but the first report on excess lung cancer appeared in 1963 by Wagoner et al.[7] In the meantime, a suggested exposure limit was established by Holaday et al.[8] in 1957, although many mines did not meet the criterion for many years. The limit was set in terms of the decay product exposure as one working level (WL), defined as "any combination of short-lived radon decay products in one liter of air that will result in the emission of 1.3×10^5 MeV of potential alpha energy." The WL month was defined as an exposure to a concentration of 1 WL for a working month of 170 h (the corresponding SI units are 1 WL = 2×10^{-5} Jm^{-3} and 1 WLM = 3.5×10^{-3} Jh m^{-3}, but they are not widely used).

RISK ESTIMATION FOR MINERS

The only data we have for risk assessment is that from lung cancer in underground miners. There are no definitive studies from population exposures, and there is no strong evidence that radon exposure causes other cancers or other diseases. There is a pattern in the miner mortality that risk is generally proportional to cumulative exposure, that lung cancer does not appear before age 40 (the normal age for appearance of lung cancer), and that there is a minimum latent period of about five years before lung cancer appears in an exposed individual.

The cancer data for the four major groups of miners mation on the miner populations, we do not have lifetime mortality risks because a large fraction of the largest groups are still alive. Extrapolating the available data to lifetime mortality has several major problems, including the following.

1. The exposure data for individual miners are poor, particularly in the early days when exposures were highest. Many of the estimates are based on measurements in the same mine at different times or in nearby mines. Other estimates are purely educated guesses attempting to extrapolate more recent data with allowance for changes in ventilation and other factors.

2. Smoking is the predominant cause of lung cancer in most of the miners. Miners were reputed to be heavy smokers, but smoking data are no better than the radon exposure data. Risk can only be estimated by comparing the subject population with controls matched for all factors except radon exposure. The Colorado miners have the most complete smoking histories, but their data are not helpful with the other groups.

Roscoe et al.[16] made a study of nonsmoking miners in Colorado. There was a definite excess of lung cancers in the group but only among those with exposures over 360 WLM. This is an interesting observation but does not fit in with the other miner epidemiological conclusions.

3. The largest variable in the assessment is the model used to extrapolate to lifetime mortality. The absolute risk model considers the excess annual risk to be uniform from any specified exposure. The constant relative risk model considers the excess risk at any age to be a fixed multiple of the lung cancer risk in the general population at that age. Neither simple model fits the data, and current models all contain modifiers. This is not surprising because the simple models have no biological base.

4. There appears to be a reduction in risk with time since cessation of exposure, particularly apparent in the Czech miners. This would mean that more recent exposures should be weighted more heavily in risk assessment. This was originally accounted for in the National Council on Radiation Protection and Measurements (NCRP) model, which allowed for an exponential risk reduction with a 20-year half-life and was confirmed by Hornung and Meinhart,[9] who estimated a 10- to 20-year period. A reduction has been included, rather awkwardly, in the BEIR IV model by decreasing the risk factor stepwise with time since exposure and attained age.

Lubin et al.[17] made a study of Chinese tin miners, many of whom were exposed when very young. There was no indication of an increased risk with these exposures; in fact, there was a slight decrease in those who started mining before age 13. However, the reduced risk was not statistically significant.

Various groups have used the miner data to estimate the risk of exposure to radon decay products, including the NCRP,[18] the BEIR IV Committee of the National Research Council,[19] and the International Commission on Radiological Protection.[20] From the same data, they have come up with lifetime risks of from 0.9 to 3.4% for exposure to 1 WLM/yr from birth. The variation is due to differences in the methods used to extrapolate the

Table 1 Lung Cancer Mortality Experience in Five Groups of Underground Miners

Group	Study period	Number of miners	Average exposure, WLM	Lung cancer deaths: Observed	Lung cancer deaths: Expected
Colorado Plateau[9]	1951–82	3 347	882	256	59
Ontario, Canada[10]	1955–81	11 076	37	87	58
Czechoslovakia[11]	1948–80	4 043	226	484	98
Malmberget, Sweden[12]	1951–76	1 292	98	51	15

present miner mortality to completion, treatment of smoking, and treatment of the effects of age at exposure. These values, as well as those adopted by the Environmental Protection Agency (EPA),[21] are shown in Table 2. More recent data tend to favor the lower risk estimates, but the numerical risk is not a settled issue.

POPULATION EXPOSURES

Atmospheric radon arises almost exclusively from the decay of radium on the surfaces of soil particles and fractured rocks. Radon formed in the soil pore space moves by diffusion and is exhaled into the free atmosphere at an average rate of about 0.4 pCi m^{-2} s^{-1} (0.015 Bq m^{-2} s^{-1}). The global release is about 100 EBq (3 GCi/yr), which results in an average outdoor radon concentration of 8 Bq m^{-3} (0.2 pCi L^{-1}) over the continents.[23]

The exhaled radon mixes upward very rapidly in the troposphere, but, in an inversion, the radon cannot disperse, which gives rise to a general diurnal cycle with the maximum concentration in the early morning and a minimum in the afternoon when vertical mixing is a maximum. The diurnal and seasonal cycles generally show variation by only a factor of about 3.

Table 2 Estimates of Lifetime Risk of Excess Lung Cancer Mortality for Exposure to 1 WLM/yr from Birth

Reference	Projection model	Excess risk, %
NCRP (1984)[18]	Modified absolute risk	0 9
ICRP (1987)[20]	Absolute risk	1.1
	Constant relative risk	1 6
NAS/NRC (1988)[19]	Modified relative risk	
	Men	3 4
	Women	1 4
EPA (1986)[21]	Constant relative risk	1.3 to 5.0[a]

[a]Puskin and Yang[22] have indicated that this range should be modified to 0.8 to 3.0.

Air entering a building from the soil under and around the foundation has a very high radon concentration. This radon cannot disperse, and the concentration rises to levels much higher than those outdoors. The floors above ground level in high-rise apartments have less soil contact and lower radon levels. Extreme ventilation can reduce levels compared with those found outdoors, but this is not a practical solution to radon reduction. Some radon enters the home from radium in building materials, radon in domestic water supplies, or even in natural gas. These sources are almost always minor contributors except in special locations.

The data on radon in homes are extensive, but the quality and relevance of many measurements are suspect for use in assessing average population exposure. Samples in basements are generally a factor of 2 or more higher than in aboveground living quarters. Most of these have also been sampled under closed-house conditions, so the data are only useful for screening potentially high concentrations. In addition, many of the measurements have been concentrated in areas where the concentration is expected to be high.

Surveys of indoor radon concentrations have been reported from a number of countries. Radon is much easier to measure than the decay products in routine analysis, and it has been accepted as a surrogate analysis. All these surveys approximate a lognormal distribution, which means that a significant number of results are 10 to 100 times the geometric mean. On the basis of their distribution, the EPA estimates that 6% of the homes in the United States are above the criterion of 4 pCi L^{-1} (150 Bq m^{-3}) and 0.06% are above 750 Bq m^{-3} (20 pCi L^{-1}). The 20-pCi L^{-1} concentration would lead to a 50-year exposure of 250 WLM, well above the exposure producing excess lung cancers in miners.

Assessing average population exposure requires a statistically planned survey covering the entire country and measuring homes with the occupants living normal lives. Such surveys have been carried out in Canada

(Letourneau et al.),[24] Germany (Schmier and Wicke[25]), and the United Kingdom (Wrixon et al.[26]), and preliminary data for the United States are available from the EPA. The rounded averages of 30, 50, 15, and 50 Bq m^{-3} (0.8, 1.4, 0.4, and 1.4 pCi L^{-1}) reflect both radon sources and differences in living style. Sweden has shown concentrations in the range of 85 to 120 Bq m^{-3} (Swedjemark and Mjones[27]) caused by the higher soil radioactivity.

Radium contamination from industrial activities has led to local problems with high indoor radon concentrations, notably in Grand Junction, Colo., and Port Hope, Ontario. Concern has also been expressed about tailings piles from uranium mills, which has led to government-financed remediation in several states. Although this can be a significant local problem directly on or very near tailings, note that the releases from the piles are a fraction of a percent of those coming from the ground naturally in the United States.

PREDICTING POPULATION RISK

Even if the risk to miners from radon exposure were firmly established, the transfer of this risk to population exposures also has problems, including the following.

1. The miners had high-level exposures during working hours for a relatively few years, approximately 10 or 15. Environmental exposures are continuous for a lifetime and are to lower levels. There is no clear evidence of what effect this might have except that it is likely that extended exposure is less hazardous.

2. The miner population was all male and of working age. The general population is made up of both sexes who are exposed from birth.

3. Although it is true that miners in various types of mines with different secondary pollutants all show a radon response, it is not certain that there is not some other common factor that changes the miners' risk. This is true of most epidemiological studies.

4. There are a number of physical and biological factors, such as breathing rate, aerosol size and concentration, etc., that might produce a different response in the two groups. It has been possible to model the alpha radiation dose to the bronchial epithelium for various conditions. In general, the calculated dose per WLM is the same for miners and the population. A panel has produced a report for the National Research Council[28] that estimates the population dose per WLM as about two-thirds that of the miners, but other investigators show ratios closer to unity (Harley and Cohen[29]).

POPULATION EPIDEMIOLOGY

There are many epidemiological studies going on that attempt to show a relationship between lung cancer and radon in the home. In response to local pressures, most of these studies involved too few homes and have little chance of success. A sound study will require a large effort involving many homes with high radon concentrations and controls with low levels.

The simplest environmental epidemiology is the ecological study where average exposures in an area are compared with cancer statistics for that same area. The largest effort of this type was undertaken by Dr. Bernard Cohen of the University of Pittsburgh who has consistently produced data for counties throughout the United States that show negative correlations.[30] (This has been very pleasing to the hormesis enthusiasts.) The weakness of ecological studies is that the actual exposures of the individuals that develop lung cancer are not known. In addition, their smoking history is unknown. Dr. Cohen has attempted to control for variables in life-style, socioeconomic factors, etc., for the individuals submitting samples for radon analysis, but this may not match with the cancer cases. This is a large data base that cannot be ignored and requires intensive study. At some point, it may well show that the miner data overestimate the population risk.

A more direct epidemiological approach is the case control study, in which lung cancer cases are selected along with the same or larger number of matched controls. An attempt is then made to estimate the radon exposures of both groups over the last 10 to 30 years by measuring concentrations in present and previous homes. This is obviously an expensive process, and the ability of the studies to show an effect has been limited by the small number of subjects (a few hundred cases and controls). Summaries of existing studies are given in the proceedings of a Department of Energy/Commission of the European Communities workshop,[31] and the proceedings of a similar meeting in 1991 should become available later.

In the case of a control study of New Jersey women, Schoenberg et al.[32] showed an increase in risk with radon exposure; however, this increase was not statistically significant. Although the study was well-planned, it was hampered by the very limited range of exposures.

Larger studies should be able to demonstrate the presence or absence of a risk from exposure to indoor radon. It is unlikely, however, that a dose–response function can be determined.

GUIDELINES

The presence of high radon levels in a home is almost always a natural phenomenon. Thus no one is liable, and any remediation required is the responsibility of the owner. Advisory bodies and regulatory agencies in the United States have been reluctant to set legally binding limits and have settled on recommendations or guidelines.

The National Council on Radiation Protection and Measurements proposed that an individual with an annual exposure greater than 2 WLM yr^{-1} should consider remediation.[33] The ICRP (Ref. 34) recommended that new housing should be kept below 200 Bq m^{-3} (about 5 pCi L^{-1}) and existing housing below 400 Bq m^{-3} (11 pCi L^{-1}), but those limits[1] are now being reexamined. The EPA guideline in their "Citizen's Guide to Radon"[21] was set at 4 pCi L^{-1} (150 Bq m^{-3}) for a home if 75% occupancy is assumed.

Because the limits for indoor radon are guidelines only, the EPA has proposed a standard for radon in public drinking-water supplies of 300 pCi L^{-1} (11 000 Bq m^{-3}). They recognize that the risk is from transfer to indoor air during domestic use of water. The 300 pCi L^{-1} in water would lead to an air concentration of 0.03 pCi L^{-1} (about 1 Bq m^{-3}), which is about 2% of the average indoor concentration. The water limit will be a regulation and not a guideline.

MEASUREMENT OF RADON AND DECAY PRODUCTS

The measurement of radon decay products in the mines has been based on alpha counting of an air filter after sampling for a few minutes. The various techniques have been described [(e.g., NCRP, 1988 (Ref. 35)]. Such methods were invaluable in assessing instantaneous miner exposure rate but did not give an integrated exposure. By the time personal monitors were developed, uranium mining in the United States was minimal.

Measurements of decay products in the home were too labor-intensive and expensive to be practical, so attention was shifted to radon. The following three passive monitors were developed and adopted by the radon industry.

1. The charcoal canister relies on radon diffusing into a charcoal-filled container over a few days followed by gamma counting of the decay products. Sampling time is limited, and the data are probably good to about a factor of 2 at average radon levels.

2. The solid-state, nuclear-track detector samples are made by diffusion through a filter into a container holding a plastic chip. Alpha radiation produces damage tracks in the plastic that can be enlarged by an alkaline etching. The number of tracks is proportional to the average radon concentration during exposure. These detectors can be exposed for up to one year, and their precision and accuracy is a result of the care and effort taken in reading the tracks.

3. The electret ion chamber samples by diffusion into a chamber holding a charged electret film. Alpha radiation in the chamber gradually discharges the electret, and the signal is the difference between the original and final electrostatic voltages. This device can also integrate exposure over a period of one year. The chamber also responds to external gamma radiation, and it is necessary to correct for this interference.

The latter two devices were patented as the Terradex TRACK-ETCH (now Landauer RAD-TRAK) detector and the E-Perm detector, respectively. These and other systems, including real-time instrumentation, are described in the NCRP report.[35] Equipment and operating costs tend to limit the use of active instruments to research studies and real-estate transactions.

When the consumer has selected a detector type, it is necessary to determine the number and location of the monitors. The EPA (in 1986) recommended placing the monitor in the basement and maintaining closed-house conditions as much as possible. This is intended to demonstrate the radon potential in the home and is part of their mandatory measurement procedure. As mentioned earlier, this concentration can be severalfold higher than that in the living areas under normal conditions. If the basement results are high, measurements should be taken in the living areas to assess exposure. An alternative is to place initially one or more detectors in the most-used living areas of the home.

The seasonal cycles of radon can also give a severalfold change in concentration. Ideally, a year-long integrating detector would provide the best information on exposure. On the other hand, waiting a year for the data is difficult, and a few charcoal canisters spaced in time or two integrating monitors in opposite seasons can give quicker results.

Expensive remediation steps should only be considered after confirming that a problem exists. At times, professionals with recording equipment might be considered, especially in real-estate transactions where expediency is necessary.

REMEDIATION

When it has been determined that the radon levels are excessive—necessarily a value judgment on the part of the home owner—remediation possibilities must be explored. These have been described adequately [(EPA (Ref. 36), NCRP (Ref. 37)], but the best approach depends on the structure itself. It is generally agreed that the radon enters the home through the foundation and other underground structures. Subslab ventilation systems almost always reduce entry adequately and can maintain low concentrations if properly maintained.

Ventilation techniques are not always fruitful and may be costly in terms of heating and air conditioning. "Do-it-yourself" sealing of cracks and other openings is generally not permanent, although sealing off a basement sump is often very effective. Professional help is usually required, and the states maintain lists of contractors considered competent to evaluate the problem and carry out remediation.

CONCLUSIONS

Exposure to radon decay products has caused excess lung cancer in underground miners. It is highly probable that cumulative indoor exposures to levels approaching those existing in the mines also will produce lung cancer. Equivalent total exposures at lower indoor concentrations may not produce proportional effects. This is not necessarily due to failure of the linearity concept but because other factors are not presently understood.

Although these problems are being resolved, it is prudent to measure radon in homes, actively reduce high-level exposures in homes, take simple steps to reduce general exposure, and observe radon-reducing measures in new construction. Because radon exposure is a natural risk, many home owners will have to make their own personal as low as reasonably achievable (ALARA) valuation, balancing the cost of remediation against a potential risk of lung cancer.

REFERENCES

1. 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, *Annals of the ICRP*, 21, Nos. 1-3, Pergamon Press, Oxford, 1991.
2. Paracelsus, *Four Treatises of Theophrastus von Hohenheim, called Paracelsus*, Part II, On the Miners' Sickness and Other Miners' Diseases, translated by H. E. Sigerist from the original 1567 publication, Johns Hopkins Press, Baltimore, 1941.
3. G. Agricola, *De Re Metallica*, 1597, translated by H. C. and L. H. Hoover, Dover Publications, New York, 1950
4. F. H. Harting and W. Hesse, Lung Cancer, the "Bergkrankheit" in the Schneeberg Mines (in German), *Vierteljahresschr Med Oeff Gesundheitswesen*, 30. 296 (1879) (also: Vol. 31: 102 and Vol 31: 313).
5. P. Ludewig and E. Lorenser, Study of the Radon Content of Mine Air in the Schneeberg Mines (In German), *Strahlentherapie*, 17: 428 (1924).
6. E. Lorenz, Radioactivity and Lung Cancer: A Critical Review of Lung Cancer in the Miners of Schneeberg and Joachimsthal, *J Natl Cancer Inst*, 5: 1-15 (1944).
7. J. K Wagoner, R. W. Miller, F. E. Lundin, Jr, J. F Fraumenim, Jr., and M E. Haij, Unusual Cancer Mortality Among a Group of Underground Metal Miners, *N Engl J Med.*, 269: 284-289 (1963).
8. D. A. Holaday, D. E. Rushing, R D. Coleman, P F Woolrich, H L. Kusnetz, and W. F. Bale, *Control of Radon and Daughters in Uranium Mines and Calculations on Biologic Effects*, U.S. Public Health Service Report 494, U.S. Government Printing Office, Washington, 1957.
9. R. W. Hornung and T. J. Meinhardt, Quantitative Risk Assessment of Lung Cancer in U S. Uranium Miners, *Health Phys*, 4: 417-430 (1987).
10. J. Muller, W. C. Wheeler, J. F. Gentleman, G Suranyi, and R. A. Kusiak, Study of Mortality of Ontario Miners, *in Proceedings of the International Conference on Occupational Safety in Mining*, H Stocker (Ed), Report CONF-841016-, pp. 335-343, Canadian Nuclear Association, Toronto, 1985.
11. J. Sevc, E. Kunz, L. Tomasek, V. Placek, and J Horacek, Cancer in Man After Exposure to Rn Daughters, *Health Phys*, 54(1): 27-46 (1988).
12. E. P. Radford and K. G. Renard, Lung Cancer in Swedish Iron Miners Exposed to Low Doses of Radon Daughters, *N Engl J Med*, 310(23): 1485-1494 (1984).
13. G. R. Howe, R. C. Nair, H. B Newcombe, A. B Miller, and J. D. Abbatt, Lung Cancer Mortality (1950–1980) in Relation to Radon Daughter Exposure in a Cohort of Workers at the Eldorado Beaverlodge Uranium Mine, *J Natl Cancer Inst*, 77 357 (1986)
14. H. I. Morrison, R. M. Semenciw, Y. Mao, and D. T. Wigle, Cancer Mortality Among a Group of Fluorspar Miners Exposed to Radon Progeny, *Am J Epidemiol*, 128(6): 1266-1275 (1988).
15. J. M. Samet, O. M. Kutvirt, R. J. Waxweiler, and C. R. Key, Uranium Mining and Lung Cancer in Navaho Men, *N Engl J Med*, 310(23): 1481-1484 (1984).
16. R. J. Roscoe, K. Steenland, W. E. Halperin, J. J. Beaumont, and R. J. Waxweiler, Lung Cancer Mortality Among Nonsmoking Uranium Miners Exposed to Radon Daughters, *JAMA*, 262(5): 629-633 (1989).
17. J. H. Lubin, Y. L. Qiao, P. R. Taylor, S. X. Yao, A. Schatzkin, B. L. Mao, J. Y Rao, X. Z. Xuan, and J. Y. Li, Quantitative Evaluation of the Radon and Lung Cancer Association in a Case Control Study of Chinese Tin Miners, *Cancer Res*, 50(1): 174-180 (1990).
18. National Council on Radiation Protection and Measurements, *Evaluation of Occupational and Environmental Exposures to Radon and Radon Daughters in the United States*, NCRP Report No. 78, 1984.
19. National Academy of Sciences, *Health Risks of Radon and Other Internally Deposited Alpha-Emitters*, NAS/NRC Report BEIR IV, National Academy Press, Washington, 1988.
20. International Commission on Radiological Protection, *Lung Cancer Risk from Indoor Exposures to Radon Daughters*, ICRP Publication 50, Pergamon Press, Oxford, 1987.
21. U.S. Environmental Protection Agency, *A Citizens Guide to Radon*, EPA Report OPA-86-004, 1986.
22. J. S. Puskin and Y. Yang, A Retrospective Look at Rn-Induced Lung Cancer Mortality from the Viewpoint of a Relative Risk Model, *Health Phys*, 54(6): 635-643 (1988)

23 National Council on Radiation Protection and Measurements, *Exposure to the Population in the United States and Canada from Natural Background Radiation*, NRCP Report No 94, 1987

24 E G Letourneau, R G McGregor, and W B Walker, Design and Interpretation of Large Surveys for Indoor Exposure to Radon Daughters, *Radiat Prot Dosim*, 7(1 4) 303-308 (1984)

25 H Schmier and A Wicke, Results from a Survey of Indoor Radon Exposure in the Federal Republic of Germany, *Sci Total Environ*, 45 307 310 (1985)

26 A D Wrixon, L Brown, K D Cliff, C M H Driscoll, B M R Green, and J C H Miles, Indoor Radiation Surveys in the UK, *Radiat Prot Dosim*, 7(1-4) 321 325 (1984)

27 G A Swedjemark and L Mjones, Radon and Radon Daughter Concentrations in Swedish Homes, *Radiat Prot Dosim*, 7 341 (1984)

28 National Research Council, *Comparative Dosimetry of Radon in Mines and Homes Panel on Dosimetric Assumptions Affecting the Application of Radon Risk Estimates* National Academy Press, Washington, 1991

29 N H Harley and B S Cohen, Updating Radon Daughter Dosimetry, in *American Chemical Society Symposium on Radon and Its Decay Products*, P K Hopke (Ed), pp 419-429, ACS, Washington, 1987

30 B L Cohen, Expected Indoor Radon Levels in Counties with Very High and Very Low Lung Cancer Rates, *Health Phys*, 57(6) 897-907 (1989)

31 Department of Energy/Commission of the European Communities, *International Workshop on Residential Radon Epidemiology Workshop Proceedings*, Report CONF 8907178, NTIS, 1989

32 J B Schoenberg, J B Klotz, H B Wilcox, G P Nicholls, M T Gil-del-Real, A Stemhagen, and T J Mason, Case-Control Study of Residential Radon and Lung Cancer Among New Jersey Women, *Cancer Res*, 50 6520 (1990)

33 National Council on Radiation Protection and Measurements, *Exposures from the Uranium Series with Emphasis on Radon and Its Daughters*, NCRP Report No 77, 1984

34 Principles for Limiting Exposure of the Public to Natural Sources of Radiation, International Commission on Radiological Protection Publication 39, *Annals of the ICRP*, 14(1) 1-8, Pergamon Press, Oxford, 1984

35 National Council on Radiation Protection and Measurements, *Measurement of Radon and Radon Daughters in Air*, Report NCRP-97, 1988

36 Environmental Protection Agency, *Application of Radon Reduction Methods* (Revised), Report EPA/625/5-88/024, EPA, 1989

37 National Council on Radiation Protection and Measurements, *Control of Radon in Houses*, Report NRCP-103, 1989

On the Importance of the Atmospheric Parameters in the Fission Products Distribution of a Severe Reactor Accident

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Abstract: *The Istanbul TRIGA Mark II reactor is situated on a hill outside the populated city area Its geographical situation is 41°06'32" north latitude and 29°01'44" east longitude In the reactor safety report and in other publications, the safety aspects of the reactor and the ^{131}I isodose distribution have already been investigated For this article, the downwind distance of the maximum concentration was investigated for different meteorological conditions of Istanbul The particle size distribution effect will also be discussed, and the importance of atmospheric parameters will be pointed out*

Some of the many contributors to uncertainty in reactor accident consequence assessment are the wind speed, the wind direction, and the height of the stack. In addition, the particle size distribution (PSD) is a major factor in uncertainty assessment. The wind conditions and the stack height are important for the determination of the maximum dose deposition distance. The maximum dose of the released radioactivity is also calculated accordingly. Current consequence models generally do not include the effect of a PSD in their calculation, despite the fact that wet and dry deposition in the environment, as well as deposition in the lung airways and hence internal exposure resulting from inhalation of radionuclides, all depend on particle size.

In the following discussion, all these factors are investigated for the hypothetical Istanbul TRIGA Mark II reactor accident. The distances of maximum dose deposition for different wind direction and speeds are calculated according to the statistics of all meteorological data concerning the reactor site. The particle size effect will also be discussed.

RELEASE OF FISSION PRODUCTS

The fission products that accumulate during the life of a reactor can be released to the environment in various ways. The abnormal incidents that should be taken into account are as follows:

- Cladding rupture
- Reactivity accident
- Loss-of-coolant accident
- Human errors
- External factors, such as sabotages or aircraft crashes

If one of these events occurs, some gaseous or solid fission products are released to the atmosphere and dispersed into the environment. An explanation of some of these incidents has been given in previous investigations.[1,2]

The most famous example of human error is the Chernobyl accident. This accident had an impact on many European countries as well as on some countries in Asia. Even though human health was not seriously affected, the economics of many countries received some negative impact from this accident.

In a study of uncertainty, external factors are of great importance. An aircraft crash that occurred on a research reactor site could produce very serious damage because of the absence of safety buildings. Likewise, sabotage has become a routine consideration, especially between unfriendly countries. With all these possibilities, a release of radioactive products does not have a zero probability of occurrence.

If one of these events does occur, a major question to be answered is how to know the maximum dose concentration and its deposition area. To study this problem for the Istanbul case, all the meteorological data available were analyzed and wind characteristics were considered (Table 1).

As shown in Table 1, the dominant wind direction at the TRIGA site is from northeast to southwest; the probability of this direction is, in the lower case, 15% in December, and in the higher case, 47% in July. With

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Table 1 Wind Characteristics for Istanbul TRIGA MARK-II Reactor Site Over 30-year Period[a]

		N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Calm	A	B
Jan	p	0.09	0.11	0.18	0.03	0.02	0.01	0.01	0.02	0.05	0.07	0.12	0.03	0.03	0.01	0.03	0.05	0.12	SSW/31.5	4.30
	v	5.2	5.8	7.7	4.6	3.6	2.0	2.3	3.1	2.4	3.7	4.0	3.3	3.0	2.7	4.3	4.9			
Feb	p	0.11	0.11	0.19	0.02	0.02	0.01	0.02	0.02	0.08	0.13	0.13	0.04	0.02	0.02	0.03	0.03	0.11	WNW/30	4.40
	v	6.2	4.6	7.1	4.6	5.0	2.8	2.6	3.2	2.5	3.9	4.3	4.1	3.0	3.4	4.1	4.6			
Mar	p	0.11	0.13	0.27	0.02	0.01	0.00	0.01	0.01	0.03	0.06	0.10	0.03	0.01	0.01	0.02	0.04	0.13	WSW/27.5	4.50
	v	5.1	5.1	7.2	5.0	3.7	2.0	1.8	1.9	2.1	3.8	3.9	4.4	2.5	2.6	4.7	4.4			
Apr	p	0.06	0.11	0.29	0.01	0.02	0.00	0.01	0.01	0.04	0.05	0.11	0.05	0.02	0.01	0.01	0.03	0.15	SW/26.4	3.80
	v	3.8	4.3	6.4	3.4	5.3	1.1	1.4	1.8	1.7	3.6	3.4	3.8	3.0	2.5	3.0	4.0			
May	p	0.08	0.12	0.32	0.02	0.03	0.01	0.01	0.01	0.03	0.03	0.06	0.03	0.01	0.01	0.02	0.02	0.19	WNW/24.3	3.40
	v	3.4	4.0	5.5	3.7	5.1	1.3	1.4	1.4	1.6	3.0	2.8	3.7	2.6	1.9	2.6	3.4			
June	p	0.09	0.11	0.38	0.04	0.04	0.01	0.01	0.01	0.02	0.01	0.04	0.02	0.01	0.01	0.02	0.02	0.16	NE/25.5	3.90
	v	4.2	3.8	6.1	3.2	4.2	1.2	1.2	1.0	1.5	2.4	3.0	3.6	2.3	2.4	2.8	3.1			
July	p	0.10	0.12	0.47	0.06	0.05	0.01	0.01	0.00	0.02	0.01	0.01	0.01	0.02	0.01	0.01	0.02	0.10	NNW/29.4	4.80
	v	4.3	4.3	6.8	5.4	4.6	1.0	1.3	1.2	1.0	1.8	1.9	3.6	2.2	2.3	3.0	4.4			
Aug	p	0.10	0.13	0.41	0.08	0.06	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.02	0.02	0.10	WNW/24.4	5.00
	v	4.4	4.7	7.1	5.3	4.3	2.8	1.0	1.1	1.4	1.8	2.2	2.5	3.5	2.2	5.3	3.4			
Sept	p	0.08	0.10	0.36	0.10	0.07	0.01	0.01	0.00	0.02	0.02	0.02	0.02	0.01	0.01	0.02	0.02	0.13	SSW/23.3	4.80
	v	4.2	4.6	7.2	6.0	5.3	1.9	1.7	1.1	1.4	2.4	2.5	3.3	2.6	2.7	4.6	3.4			
Oct	p	0.07	0.12	0.29	0.07	0.04	0.01	0.01	0.01	0.03	0.03	0.02	0.03	0.02	0.01	0.02	0.01	0.15	NE/27.7	4.40
	v	4.5	5.5	7.2	6.0	4.8	1.6	1.1	1.6	1.9	2.7	2.6	3.2	2.9	2.5	3.9	3.6			
Nov	p	0.07	0.08	0.21	0.04	0.02	0.01	0.02	0.01	0.05	0.07	0.12	0.06	0.03	0.01	0.02	0.02	0.14	NNE/28.4	3.70
	v	4.3	4.5	6.4	4.8	2.9	1.1	1.8	1.7	2.8	3.9	3.3	3.7	2.9	2.4	4.1	4.3			
Dec	p	0.09	0.08	0.15	0.04	0.02	0.01	0.02	0.02	0.06	0.08	0.12	0.04	0.02	0.02	0.02	0.04	0.14	WSW/27.6	3.90
	v	4.8	5.2	6.9	6.1	3.2	1.8	1.9	3.1	2.7	4.3	3.6	3.1	3.0	2.7	4.3	5.2			
Yearly	p	0.09	0.11	0.29	0.05	0.03	0.01	0.01	0.01	0.03	0.04	0.08	0.03	0.02	0.01	0.02	0.03	0.14	SSW/31.5	4.20
average	v	4.6	4.7	6.7	5.2	4.5	1.7	1.7	2.3	2.1	3.6	3.5	3.6	2.8	2.6	4.0	4.3			

[a] p, occurring probability, v, average wind speed at 10 m from ground level (m/s), A, direction and value of the maximum wind speed, B, weighted average of the speed; N, north, S, south, E, east; W, west

this realistic data, the investigation was carried out in two areas: the diffusion of gaseous fission products and the dry deposition of fission particles.

THE DIFFUSION THEORY

For the investigation of the diffusion of gaseous products, the following well-known concentration equation was used:[3]

$$C = \frac{2 \cdot Q \cdot x^{n-2}}{\pi \cdot C_y \cdot C_z \cdot u} \exp(x^{n-2}) \left(\frac{y^2}{C_y^2} + \frac{h^2}{C_z^2} \right) \quad (1)$$

where C = radioactivity at the point x
Q = activity released from the stack of height h
C_z = diffusion coefficient
C_y = diffusion coefficient
u = average wind speed at the exit of the stack
n = stability coefficient

Note from Eq. 1 that $C \rightarrow 0$ as $x \rightarrow 0$ (that is, as the base of the stack is approached). Thus the maximum ground concentration (C_m) occurs at some distance downwind of the source. The maximum concentration (C_m), as well as its downwind distance (x_m), may be obtained by maximizing Eq. 1 with respect to x, which leads to

$$C_m = \frac{2 \cdot Q}{e\ \pi \cdot u \cdot h^2} \frac{C_z}{C_y} \quad (2)$$

$$x_m = \frac{h}{C_z} \frac{2}{2-n} \quad (3)$$

According to these equations, x_m and C_m were calculated with respect to Q, h, and n values. For the variation of C_m, the preceding equation was put in the following form:[4]

$$C_m = K\ Q \quad (4)$$

In this equation, C_z was assumed to be equal to C_y, and the K factor is defined by

$$K = \frac{2}{e \cdot u \cdot \pi \cdot h^2} = \frac{0.234}{uh^2} \quad (5)$$

Generally, wind speeds are measured at 10 m from ground level. To obtain the values of these wind speeds at the height h, the following equation is used with respect to the stability constant of the atmosphere:

$$u_h = u_{10}\left(\frac{h}{10}\right)^n \quad (6)$$

In the calculation of C_m, an angular distribution of α = 10° for wind blow was assumed.[4] With these assumptions, the K factor can be calculated for different h and n values. The results are reported in Fig. 1.

Tables 2 and 3 give the downwind distances for different situations. Table 2 analyzes the situation with respect to the wind directions for h = 25 m and n = 0.25. In Table 3, only maximum and minimum downwind distances of the maximum concentration are reported for a different stack height and stability constant.

Table 2 shows that the minimum downwind distance from the reactor is 458 m with an east-southeast wind in July, and the maximum distance is 622 m with a northeast wind in January. If one takes into account the yearly weighted average, these distances are 498 and 605 m. But, as it is easy to see from Table 1, the dominant wind for the reactor site is from the northeast direction; so the distance of 605 m can be taken as the distance where maximum dose rate will be collapsed. This value is also in good agreement with the value obtained in Ref. 1. To know the radioactivity deposition in this place, one has to know the Q value of the gaseous radioisotopes and make calculations with the values of K obtained from Fig. 1.

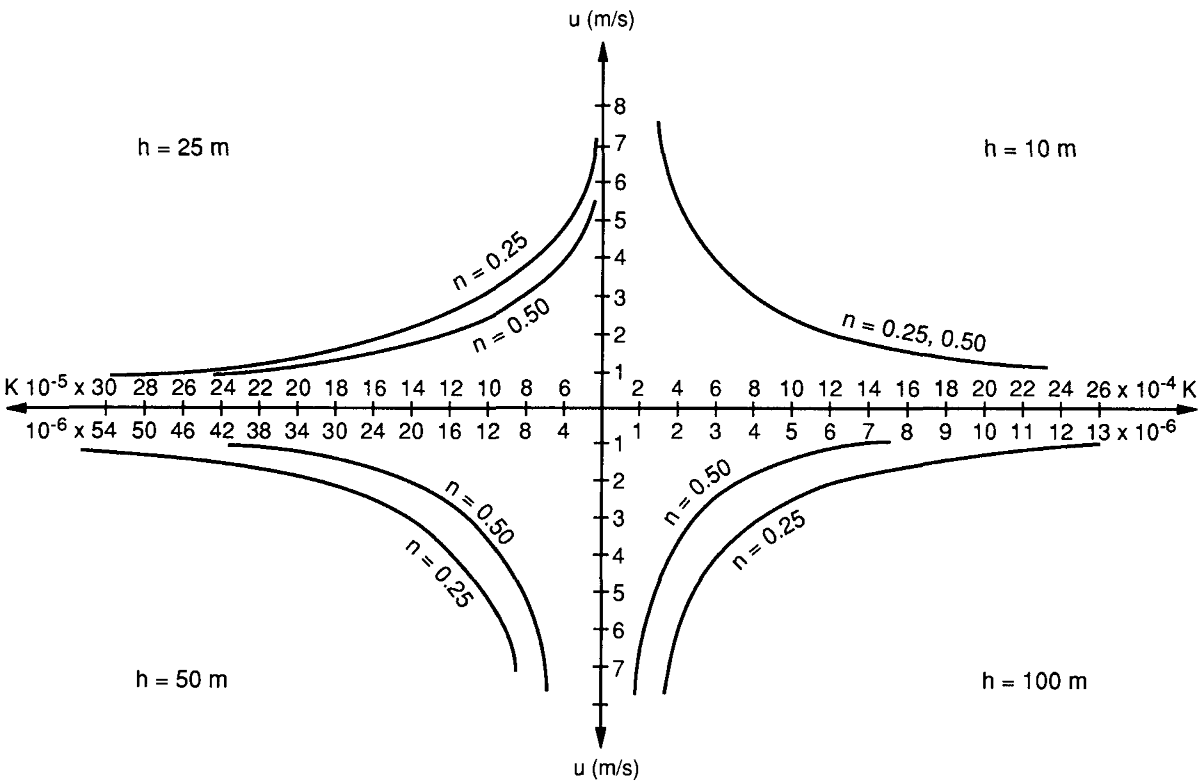

Fig. 1 K values with respect to h and n.

Table 2 The Maximum Concentration Downwind Distances (x_m) for h = 25 m and n = 0.25 (m)[a]

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Average	Maximum
Jan	588	598	622	578	558	513	524	546	527	560	567	551	544	536	572	583	572	761
Feb	603	578	615	578	585	538	533	549	530	564	572	568	544	553	568	578	574	755
Mar	586	586	616	584	560	513	505	509	516	562	564	574	529	532	579	574	576	745
Apr	560	570	603	551	587	469	485	503	499	555	551	560	541	527	541	564	560	738
May	549	561	588	555	581	478	483	483	493	539	534	555	528	505	528	549	549	727
June	563	555	594	542	563	471	471	459	486	520	537	551	517	520	531	539	557	729
July	564	564	602	582	569	458	475	470	458	498	502	550	512	516	531	566	573	742
Aug	566	571	606	581	564	530	458	464	480	498	512	522	547	512	581	545	576	722
Sept	563	571	608	593	582	503	495	465	481	520	523	544	526	529	571	547	574	719
Oct	571	587	611	595	576	492	467	492	505	531	528	544	536	525	559	553	569	740
Nov	569	573	602	578	538	468	502	498	535	561	548	557	538	523	565	569	557	745
Dec	580	586	611	600	547	504	508	545	534	571	556	545	542	534	571	586	563	744
Yearly average	574	575	605	584	572	498	498	520	513	554	552	554	534	529	562	586	566	761

[a] N, north, S, south, E, east, W, west

Table 3 The Maximum and Minimum Downwind Distances

		n = 0.25				n = 0.50			
		h = 10 m	h = 25 m	h = 50 m	h = 100 m	h = 10 m	h = 25 m	h = 50 m	h = 100 m
Jan	max	211	622	1 408	3 188	1 816	7 179	20 305	57 428
	min	174	513	1 162	2 629	1 159	4 580	12 955	36 641
Feb	max	209	615	1 392	3 150	1 766	6 982	19 748	55 856
	min	180	530	1 199	2 714	1 247	4 930	13 945	39 442
Mar	max	209	616	1 393	3 154	1 771	7 001	19 803	56 010
	min	171	505	1 143	2 587	1 116	4 411	12 475	35 284
Apr	max	205	603	1 365	3 090	1 688	6 674	18 876	53 390
	min	159	469	1 061	2 403	939	3 711	10 495	29 684
May	max	200	588	1 330	3 011	1 589	6 282	17 767	50 254
	min	162	478	1 082	2 450	983	3 884	10 985	31 072
June	max	202	594	1 345	3 043	1 630	6 442	18 221	51 538
	min	156	459	1 038	2 350	892	3 526	9 921	28 207
July	max	204	602	1 363	3 084	1 681	6 645	18 975	53 162
	min	155	458	1 036	2 345	887	3 508	9 921	28 060
Aug	max	206	606	1 371	3 103	1 705	6 739	19 060	53 909
	min	155	458	1 036	2 345	887	3 506	9 917	28 048
Sept	max	207	608	1 377	3 117	1 723	6 811	19 265	54 490
	min	158	465	1 053	2 383	921	3 641	10 299	29 130
Oct	max	207	611	1 382	3 128	1 738	6 869	19 428	54 950
	min	159	467	1 057	2 392	929	3 672	10 386	29 375
Nov	max	204	602	1 363	3 085	1 682	6 649	18 807	53 194
	min	159	468	1 060	2 399	935	3 697	10 457	29 575
Dec	max	207	611	1 382	3 128	1 738	6 870	19 432	54 962
	min	171	504	1 141	2 582	1 111	4 390	12 416	35 119

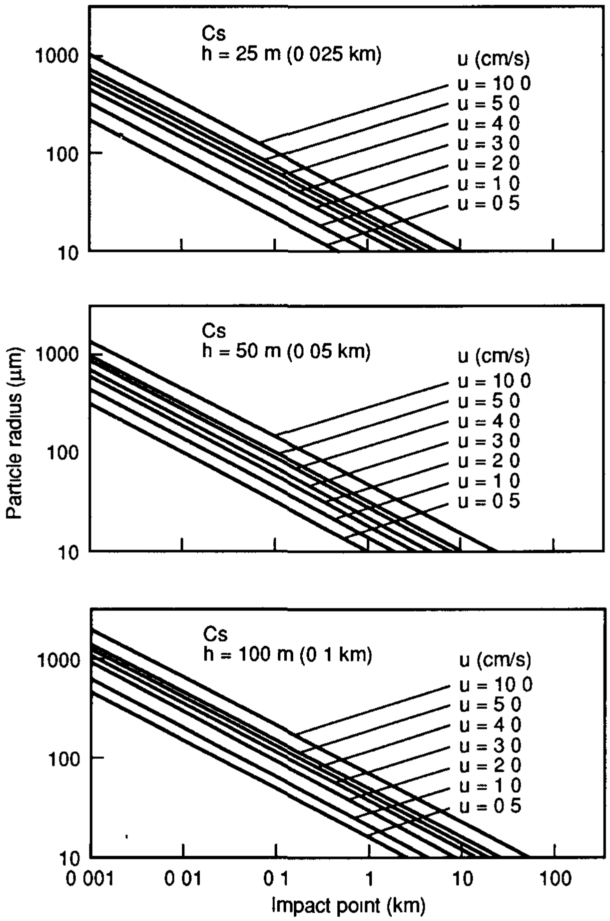

Fig. 2 Maximum concentration distances vs. cesium particle size.

DRY DEPOSITION OF FISSION PARTICLES

When the fission products are not in gaseous form, but are small particles, the diffusion theory is not applicable in all cases. For small particles of some micrometers in radius, the maximum concentration and its downwind distance *(X_m)* can also be calculated by using the diffusion deposition equations. For particles 60 µm or more in radius, the diffusion theory is not admitted as reliable, and the dry deposition velocity is then calculated according to the gravity forces. This deposition velocity will then permit calculation of the impact point of the particles.

This distance x of a particle of radius r can be calculated by using the following formula :

$$x = 0.812 \cdot 10^{-6} \cdot \frac{u \cdot h}{\delta \cdot r^2} (1.002)^{t-10} \qquad (7)$$

where u = wind speed at $\frac{2}{5}$ of the stack height, cm/s
h = stack height, cm
δ = density of the fission-product particle, g/cm^3
r = particle radius, cm
t = atmospheric temperature, °C

In this equation, the coefficient was calculated according to the viscosity of the atmosphere at 10°C.

For different stack heights, the downwind distances of the maximum concentration are calculated with respect to the particle radius. The results are plotted in Fig. 2 for cesium (g = 1.873 g/cm^3) and in Fig. 3 for plutonium (g = 19.6 g/cm^3).

CONCLUSIONS

Generally, the diffusion theory is adopted for the safety calculations of nuclear reactors, and, in these

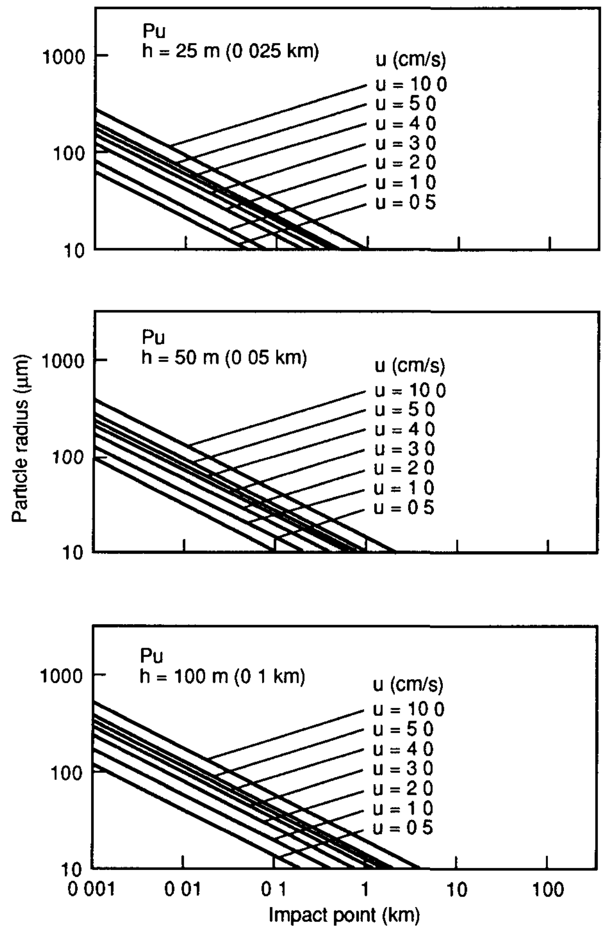

Fig. 3 Maximum concentration distances vs. plutonium particle size.

calculations, I-131 is almost always taken as an example. But the particle release from a reactor accident is as important as the gaseous fission products. The most spectacular example of this release is the cesium and strontium distribution from the Chernobyl accident. The deposition of these particles, among various others, on tea leaves contributed to the radioactive pollution of Turkish tea. Around 50 000 tons of tea leaves are waiting for a final deposition area.

In this investigation, one can see the importance of this factor, as illustrated in Figs. 2 and 3. If the particle radius of the released fission product is known, then the impact point of this isotope can be calculated from a similar figure with respect to the wind speed and direction. The diffusion theory can also be used for the maximum concentration calculations of particles with up to 60-μm radius.[5] In all these calculations, the particles were assumed to be spherical. For a more realistic investigation, a roughness factor must be included in the equations.

Also note that, for these investigations, a pure fission-product particle was assumed. But if a small fission-product particle were adhered onto a bigger dust particle, these calculations and distances are not true. In that case, one would have to know the density of this carrying particle (e.g., soil or concrete particle) to be able to calculate the deposition distance.

From these investigations, Fig. 4 was obtained, where it is easy to see the variation of x_m distance with the wind speed, with respect to the n factor, according to the diffusion theory. In this figure the stack height was taken as 25 m. Figure 5 shows the same variation for a stack height of 100 m.

According to Fig. 4, for the most probable wind direction (northeast), the impact point of the maximum concentration will be at 6 600 m far from the reactor in July, for $n = 0.50$. But for a stack height of 100 m for the same ground velocity of 6.8 m/s in July, this distance will be 53 000 m.

As can be seen from this investigation, for $n = 0.20$ or $n = 0.25$, the downwind distance of the maximum concentration is not very long and is situated in the campus of the University, but for $n = 0.50$ and for a high stack, the populated area of the city of Istanbul will receive some amount of radioactivity with the most probable wind direction (northeast).

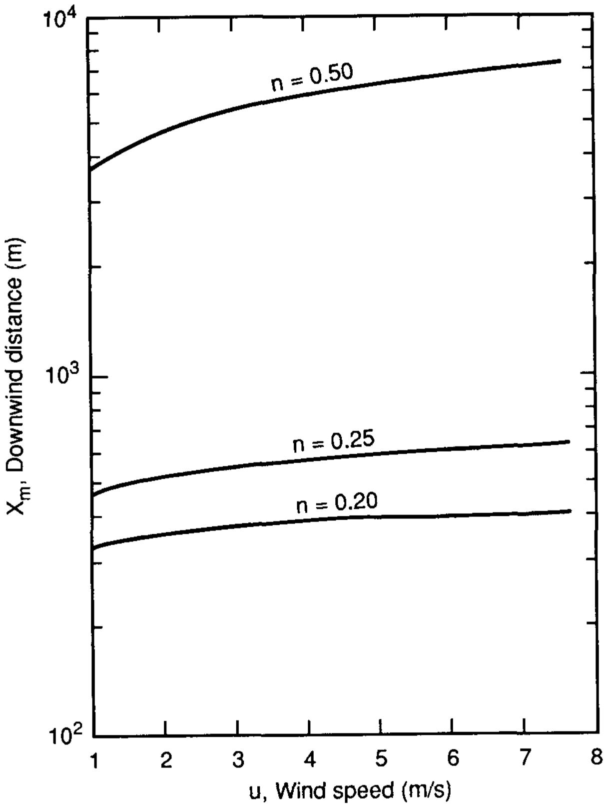

Fig. 4 Maximum concentration distances vs. wind speed for stack height of 25 m.

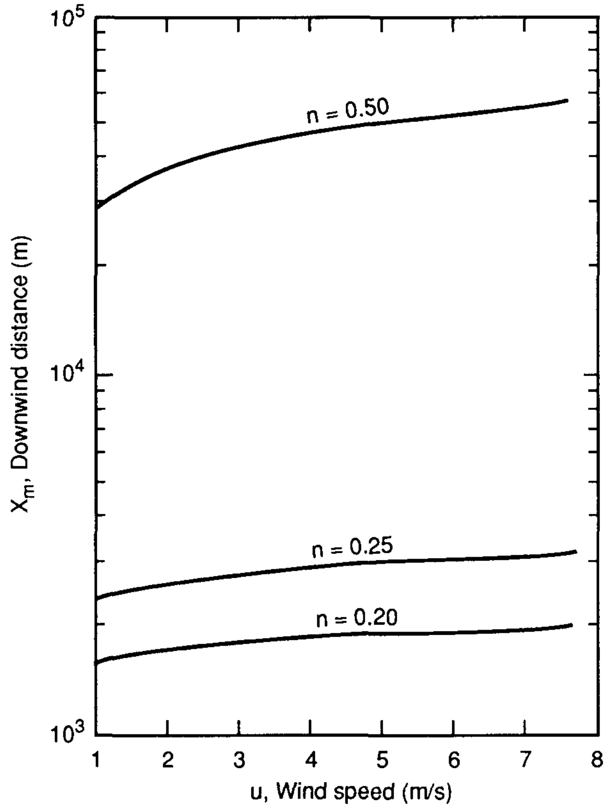

Fig. 5 Maximum concentration distances vs. wind speed for stack height of 100 m.

The tables and figures show also that, with the southwest wind, the danger of radioactive pollution of the populated area has no importance because the same distances extend to the Black Sea zone where only environmental pollution must be taken into account.

As a general conclusion of this investigation, we can say that the most important parameters in the distribution of fission products are the stack heights and the stability coefficient of the atmosphere. If the stability coefficient increases to 100%, the downwind distance (x_m) increases to 1025%. The same increase in the stack height increases this distance to 133%. For the wind speed, the same increase is only 15%. It is easy to see from these results that the stability of the atmosphere is a very important factor in fission-product distribution.

Finally, we have to point out the importance of this kind of investigation. In areas where a nuclear plant, a research reactor, or a power reactor are built, emergency planning and organizations are strongly recommended. The security authority must know the radioactivity distribution for a reactor accident. This investigation had the aim of obtaining this knowledge for the TRIGA reactor.

REFERENCES

1 N Aybers and H Yavuz, About the Safety Analysıs of Istanbul TRIGA Mark-II Reactor, *V European TRIGA Users Conference, September 4-6, 1978, Portoroz.*

2 General Atomıc, *Safety Analysıs Report for the TRIGA Mark-II Reactor for the Technıcal Unıversıty of Istanbul*, Report E-117-478, October 1975

3 O G Sutton, *Mıcrometeorology: A Study of Physıcal Processes ın the Lowest Layers of the Earth's Atmosphere*, McGraw-Hıll Book Company, 1953

4 S. Oney, Reaktor Kazalarında Radyoaktıf Baca Gazlarının Atmosfer Kosullarına Baglı Dıfuzyonu ve Enversıyon Kosullarının Incelenmesı, Ph D Thesıs, Istanbul, 1973

5. *Smıthsonıan Meteorologıcal Tables*, Smıthsonıan Instıtutıon Publıcatıon 4014, 1951

Book Review: *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V*

By C. R. Richmond[a]

After a mighty struggle, the National Academy of Sciences' Committee on the Biological Effects of Ionizing Radiation (BEIR) has published the 1990 report: *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V* (421 pages). It is the sixth in a series of reports. The first report, *Biological Effects of Atomic Radiation*, published in 1956, is sometimes referred to as the BEAR report.[1] The titles of BEIRs I-V refer to ionizing rather than atomic radiation.[2–6] Some of these reports have been controversial.

BEIRs II and IV dealt with specific subjects. For example, BEIR IV dealt with high linear-energy-transfer (LET) radiations. The other reports considered all forms of radiation, especially those of low LET. Basically, BEIRs I, III, and V represent analyses of the biological effects on the atomic bombing survivors, incorporating progressively longer periods of time following the initial exposure, changes in dosimetry, changes in the various dose–response, and cancer projection models used by the committees. Limited information obtained from other groups of exposed human subjects, usually patients exposed to high radiation doses and high dose rates and data from animal studies, is included.

BEIR V recommendations, like those of its predecessors, are heavily dependent on data obtained by the Radiation Effects Research Foundation (RERF)[b] from survivors of the 1945 atomic bombings at Hiroshima and Nagasaki who were acutely exposed to high dose rate and primarily high-dose radiation. Not treated are other populations, such as those living in what are commonly referred to as "high background areas" in India (Kerala State) and the People's Republic of China (Guangdong Province). Health effects information, including life span, obtained from these individuals may be more relevant to current radiation protection situations as the exposures are protracted and both the radiation dose and dose rate are low (multiples of natural background). Also untouched are the large populations involved in major nuclear accidents, such as the releases into the Techa River and from the Kyshtym nuclear complex in the Cheliabinsk Province of the Soviet Union. Members of these populations received both low and high radiation doses.

Many people interested in radiation protection will read only the executive summary or perhaps selected parts of BEIR V. Few will care to read the entire report and fewer still would understand the entire report. What may not be apparent to the casual reader is the limited selection of data upon which the conclusions are based.

Perhaps the key message of BEIR V is that the assigned risk from radiation is higher than previously estimated in BEIR III by a factor of about 3 or 4. How did the committee arrive at this conclusion? Three major factors contribute to the revision:

1. Revised estimates of radiation dose received by the atomic bombing survivors.[c]
2. Additional elapsed time since BEIR III for the annual excess cancer mortality to be expressed.[d]
3. Use of different risk-projection and dose–response models—particularly a linear, rather than a linear-quadratic, function to relate nonleukemic (solid) cancers and radiation dose.[e]

Use of the DS86[f] dosimetry decreased the average organ dose by a factor of about 2 from BEIR III estimates. Earlier health effects studies of the atomic bombing survivors by BEIR committees were based on T65D[g] dosimetry.

[a]Oak Ridge National Laboratory

[b]Hiroshima, Japan. Formerly the Atomic Bomb Casualty Commission Managed for the U S. Department of Energy by the National Academy of Sciences

[c]The revised dosimetry decreased the average organ dose from earlier estimates based on 1965 dosimetry

[d]BEIR V includes data analyzed through 1985 for atomic bombing survivors, BEIR III used information available up to about 1975.

[e]The BEIR V committee believes the use of the constant additive (absolute) risk model is no longer tenable. Thus values are given only for analyses utilizing the multiplicable (relative) risk model analyses BEIR V recommends risk values for leukemia based on dose equivalents less than 4 Sv to bone marrow

[f]DS86 is the dosimetry system produced by the RERF in 1986

[g]T65D is the tentative dosimetry system produced in 1965 by Auxier et al.

For solid cancers, the BEIR V committee selected a linear dose–response model over the linear-quadratic model used in BEIR III. The committee recognized that the risks at low doses and low dose rates might be less than predicted by the linear model. The committee stated that

> On the basis of available evidence, the population-weighted average lifetime excess risk of death from cancer following an acute dose equivalent to all body organs of 0.1 Sv (0.1 Gy of low LET radiation) is estimated to be 0.8%, although the lifetime risk varies considerably with age at the time of exposure. For low LET radiation, accumulation of the same total dose over weeks or months, however, is expected to reduce the lifetime risk appreciably, possibly by a factor of 2 or more.[6]

A. C. Upton, chairman of the BEIR V committee, reported the following in *Physics Today*:[7]

> . . . the new estimates do not differ greatly from those that the United Nations Scientific Committee on the Effects of Atomic Radiation and the BEIR I committee derived through the use of risk models analogous to the one used by the BEIR V group.

Upton[7] points out that the lifetime excess cancer mortality attributable to the acute radiation dose of 1 Gy of low-LET irradiation to the whole body is about the same as that reported in BEIR I when the multiplicative (relative) risk projection model is used. These values are 620 cancer mortalities per 10^5 persons (BEIR I) and 885 cancer mortalities per 10^5 persons (BEIR V). For the same conditions, BEIR III reported a range of 230 to 500 cancer mortalities per 10^5 persons. Of course, these should all be projected cancer mortalities. Recall that the publication dates for BEIRs I, III, and V were 1972, 1980, and 1990, respectively.

In its 1988 report, the United Nations Committee on the Effects of Atomic Radiation[8] gave a range of 700 to 1100 excess lifetime cancer mortalities per 10^5 persons following 1 Gy of acute whole-body low-LET irradiation. The agreement with BEIR V should not be surprising because both groups used essentially the same data for analysis.

Although the committee recognized the possible existence of a dose-rate effectiveness factor (DREF) of 2 to 10, a specific value was not used or recommended in BEIR V. The rationale escapes me. If asked to pick a number between 2 and 10, I doubt if most people would choose zero! Actually, the BEIR V report implies a DREF of about 4 for nonleukemic cancers. The committee wrote

> The BEIR III Committee's linear-quadratic dose–response model for solid cancers, unlike this Committee's linear model, contained an implicit dose rate factor of nearly 2.5; if this factor is taken into account, the relative risk projections for cancers other than leukemia by the two committees differ only by a factor of about 2 (Ref. 6).

There is no need to apply a DREF to the leukemia risk factors because an implicit DREF value of about 2.5 has been used in the risk projection model. The best single estimate for the human leukemia DREF derived in BEIR V is 2.1.

The BEIR V committee also addressed the question of health effects observed in irradiated populations other than the atomic bombing survivors. They wrote,

> Carcinogenic effects of radiation on the bone marrow, breast, thyroid gland, lung, stomach, colon, ovary, and other organs reported for A-bomb survivors are similar to findings reported for other irradiated human populations. With few exceptions, however, the effects have been observed only at relatively high doses and high dose rates. Studies of populations chronically exposed to low-level radiation, such as those residing in regions of elevated natural background radiation, have not shown consistent or conclusive evidence of an associated increase in the risk of cancer.[6]

On the basis of a review of populations exposed to low-level occupational or environmental radiation, the committee reported that

> . . . the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out. At such low doses and dose rates, it must be acknowledged that the lower limit of the range of uncertainty in the risks estimates extends to zero.[6]

A word should be said about the quality factor (Q) value used in BEIR V for neutrons. Although none is recommended, the committee used a value of 20 in keeping with recommendations made by NCRP and ICRP. A value of 27.8 was used in BEIR III for the neutron Q factor.

What else does BEIR V conclude? The committee points out that, by extrapolating from mouse to man, it is estimated that at least 1 Gy[a] of low-dose-rate, low-LET radiation is required to double the mutation rate in man. They also remind us that "Heritable effects of radiation have yet to be clearly demonstrated in man . . . ."[6]

BEIR V estimates one to two excess cases of genetic effects per million live births per 0.01 Sv per generation (30 years). The estimate is highly uncertain yet comparable to values found in the BEIR III and 1988 UNSCEAR reports.

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[a] $1\ \text{Gy} = 1\text{J} \times \text{kg}^{-1} = 100$ rad.

Risk of severe mental retardation following *in utero* exposure is given in BEIR V as about 43% per 1 Gy if exposure occurs during the 8th to 15th week of gestation. This risk is reduced for *in utero* exposures at other gestational ages. In addition, the committee suggests "that a threshold for the effect may exist in the range 0.2 to 0.4 Gy" (Ref. 6).

One must also exercise caution in interpreting the cancer mortality data used in BEIR V. Recent data by Shimizu et al.[9] suggest that there is no difference in cancer mortality between the control population and atomic bombing survivors receiving more than 0.2 and less than 0.5 Gy. One must also appreciate the relatively small number of observed cancers for some specific tissues and that more cancers are observed among the females in the survivors.

Because the cancer mortality figures derived from the Japanese atomic bombing survivors have been used heavily to derive new risk estimates in BEIR V, it is relevant to ask about the average life span of these individuals. Relatively little has been done by those who analyze the RERF data with regard to the life-span shortening of the survivors. Considerable emphasis has been placed on the numbers of leukemias and solid cancers that have developed, dosimetry, dose–response functions, mortality projection models, and basic biological studies. It would seem rather important to consider life span as well. This approach is rather simple and could tell us much about what may be the ultimate health effect of the radiation exposure. I am certain there are many arguments against using life span as the indicator of health effects. The question, I believe, is why should we ignore this easy approach once we have the necessary data from the RERF and other data sources.

A key unanswered question needs to be addressed. That is, has there been life-shortening for the atomic bombing survivors when compared with the control population. Perhaps of more importance, is there any evidence for life-shortening for the subpopulations receiving radiation doses of, say, less than 0.2 or 0.5 Sv.

BEIR reports are used as technical inputs to federal agencies responsible for developing operational or regulatory guides and standards. It is therefore of interest to see how agencies have responded to BEIR V. In 1991 the Department of Energy (DOE) published a report based on a technical evaluation of the BEIR V recommendations.[10] These experts advised that DOE should not incorporate BEIR V recommendations into its practices and that DOE should wait until the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) publish revised radiation protection guidance. ICRP's revisions of its basic radiation protection recommendations were published in 1991 as ICRP report No. 60 (Ref. 11). The previous ICRP recommendations were published in 1977 (Ref. 12).

The technical representatives of the Committee for Interagency Radiation Research and Policy Coordination (CIRRPC) are presently analyzing BEIR V and the 1988 UNSCEAR report. CIRRPC will produce a statement on behalf of its member agencies. Incidentally, CIRRPC has not yet accepted the neutron Q factor of 20 recommended by NCRP and ICRP and used by BEIR V.

One must be cautious in applying the BEIR V risk estimates to members of the nuclear work force. Lapp[13] argues very convincingly that

> . . . the BEIR V risk assessment increase of about 350% dwindles to about 70% when applied to the nuclear workforce exposure. Nothing has really happened that would lead to a tightening of radiation controls for a U.S. workforce whose lifetime radiation exposure averages about 5% above that to which all Americans are exposed.

BEIR V and other technical reports are reviewed by such organizations as NCRP and ICRP in developing their recommendations, which, in turn, are used by regulatory bodies, such as the U.S. Environmental Protection Agency (EPA). EPA, in turn, recommends radiation protection guidance to federal agencies. The 1977 ICRP recommendations[12] provided much of the substance for the EPA radiation protection guidance released in 1987 (Ref. 14). Despite the long interval between the two reports for review and analysis, U.S. agencies have been slow to adopt and implement the EPA guidance. In fact, DOE may have been the only agency to do so.

Most of the atomic bombing survivors are still alive, and they should be studied. However, other populations chronically exposed to low dose rates and low doses of radiation should also be used to derive estimates of risk. The BEIR V committee recognizes the importance of this by stating,

> The reported follow-up of atomic bomb survivors have been essential to the preparation of this report. Nevertheless, it is only one study with specific characteristics, and other large studies are needed to verify current risk estimates.[6]

We shall see.

The title of BEIR V is specific as to health effects of exposure to low levels of ionizing radiation. Let us all hope that more representative data are reviewed and used in BEIR VI. The use of such data may also resolve another problem: how to structure a nonredundant title for the report.

Does BEIR V add much to our knowledge of health effects of exposure to low levels of ionizing radiation? I think not. Perhaps a more accurate summary of BEIR V could have been that, considering the overall uncertainties, little evidence has accumulated since BEIR III that would argue for a change in the risk factors at this time Unfortunately, the die has been cast. BEIR V has been used and amplified by NCRP and ICRP and will probably be used by other organizations, such as EPA, as the basis for recommending an increase in radiation risk factors and therefore a lowering of the standards. If so, more money will be needed to comply with the reduced standards despite the low exposures now experienced in the nuclear industry Increasingly, money is becoming a scarce commodity and may not be as readily available as in the past. Perhaps of more consequence is the potential impact of BEIR V on the general public, whose response will doubtless be that radiation is not only bad but also worse than the "experts" previously thought

REFERENCES

1 National Academy of Sciences, National Research Council, Advisory Committee on the Biological Effects of Ionizing Radiation, *The Biological Effects of Atomic Radiation A Report to the Public* Report NP-6016, 1956

2 National Academy of Sciences, National Research Council, Advisory Committee on the Biological Effects of Ionizing Radiation, *BEIR I, The Effects on Populations of Exposure to Low Levels of Ionizing Radiations*, 1972

3 National Academy of Sciences, National Research Council, Advisory Committee on the Biological Effects of Ionizing Radiation, *Consideration of Health Benefit–Cost Analysis for Activities Involving Ionizing Radiation Exposure and Alternatives BEIR II*, 1977

4 National Academy of Sciences, National Research Council, Advisory Committee on the Biological Effects of Ionizing Radiation, *The Effects on Populations of Exposure to Low Levels of Ionizing Radiation BEIR III*, 1980

5 National Academy of Sciences, National Research Council, Advisory Committee on the Biological Effects of Ionizing Radiation, *Health Risks of Radon and Other Internally Deposited Alpha Emitters BEIR IV*, 1988

6 National Academy of Sciences, National Research Council, Advisory Committee on the Biological Effects of Ionizing Radiation, *Health Effects of Exposure to Low Levels of Ionizing Radiation BEIR V*, 1990

7 A C Upton, Health Effects of Low-level Ionizing Radiation, *Phys Today*, 44(8) 34-39, Part 1 (1991)

8 United Nations Scientific Committee on the Biological Effects of Atomic Radiation, *Sources Effects and Risks of Ionizing Radiation*, Report to the General Assembly, United Nations, New York, 1988

9 Y Shimizu, H Kato, and W J Schull, Studies of the Mortality of A-Bomb Survivors, 9 Mortality, 1950-85 Part 2, Cancer Mortality Based on the Recently Revised Doses (DS 86), *Radiat Res*, 121(2) 120-141 (1990)

10 A A Weadock, Final Report to the Secretary of Energy, *Implications of the Biological Effects of Ionizing Radiation (BEIR) V Report to the Department of Energy*, DOE Report DOE/EH 0158T, U S Department of Energy, 1990

11 International Commission on Radiological Protection, 1990 Recommendations of the International Commission on Radiological Protection, Pergamon Press, ICRP Publication 60, *Annals of the ICRP*, 21(1-3) vii & 201 (1991)

12 International Commission on Radiological Protection, Recommendations of the International Commission on Radiological Protection, Pergamon Press, ICRP Publication 26, *Annals of the ICRP*, 1(3) 53 (1977)

13 R E Lapp, BEIR V Implications for the Nuclear Workforce, *Science* 247 620-621 (1990)

14 Environmental Protection Agency, Radiation Protection Guidance to Federal Agencies for Occupational Exposure, *Federal Regist* 52(17) 2822-2834 (January 27, 1987)

Waste and Spent Fuel Management

Edited by K. J. Notz, Jr.

Activities Related to Waste and Spent Fuel Management

Compiled by M. D. Muhlheim[a] and E. G. Silver[a]

This feature includes brief reports on administrative, regulatory, and technical activities related to research, development for, and implementation of facilities and technologies related to safety aspects of the management of radioactive wastes and spent nuclear fuel.

The information in this issue of *Nuclear Safety* was received during April, May, and June of 1991.

WIPP INCHES CLOSER TO BECOMING A REALITY

The Waste Isolation Pilot Plant (WIPP) near Carlsbad, N. Mex., came a significant step closer to operational readiness with the approval, by Secretary of Energy J. D. Watkins, of revision seven of the draft decision plan for WIPP (Ref. 1). As discussed in our previous issue,[2] the WIPP facility would be the nation's first geologic repository for defense-related nuclear wastes. Plutonium-contaminated wastes from U.S. defense facilities are planned to be permanently stored 2150 ft below the surface of a 16-acre plot of desert in a salt bed that has remained "stable and free of groundwater for 225 million years." The facility, which has already cost more than $800 million to build, is intended to hold nearly one million barrels of radioactive plutonium waste from nuclear weapons plants in ten states. The draft decision plan is used by the Department of Energy (DOE) as an outline for tracking and managing the "prerequisite activities" that must precede a decision to declare WIPP's readiness to receive transuranic waste on a test basis.

[a]Oak Ridge National Laboratory

In anticipation of fulfilling the prerequisites defined in the draft decision plan, preparations were under way to begin bin loading demonstrations at the Idaho National Engineering Laboratory (INEL). Transuranic waste currently stored at INEL was to be loaded into six drums in a simulated transportation exercise that should provide DOE with valuable experience in the safe loading of waste. After the test, and following a sampling and analysis period, the actual bin loading will take place at the Argonne National Laboratory-West facility, also located in Idaho.

If the test bin loading proceeds as planned, several more bins will be loaded and subjected to verification of regulatory and WIPP acceptance requirements. If the acceptance criteria are met and authorization is granted by Watkins, the bins will be sent to WIPP. This step would mark the long awaited inception of the WIPP test phase.

The WIPP, however, cannot accept nuclear waste until the land is transferred from the Department of the Interior (DOI) to DOE. In January 1991 the Bush Administration imposed an administrative land transfer that could allow the WIPP site to open before Congress authorizes the transfer of ownership of WIPP from DOI to DOE. Toward that end, the House Committee on Interior and Insular Affairs' Subcommittee on Energy and the Environment approved by voice vote in mid-June 1991 the transfer of ownership of WIPP from DOI to DOE (Ref. 3) with a full committee markup session in late June 1991.

During the markup session, Sub-committee Chairman P. H. Kostmayer (D-Pa.) pointed out that many longtime observers of the project feel DOE has yet to make the case that WIPP is the right place to license as the nation's first underground repository for nuclear waste since potential problems with the evolution of radioactive and perhaps explosive gases from the depository have not been fully resolved.

The New Mexico delegation vowed to fight the land transfer unless the project was in full compliance with Environmental Protection Agency (EPA) standards and enough money was appropriated for emergency preparedness measures.

At the markup session, Rep. W. Richardson (D-N. Mex.) introduced an amendment that would prohibit transfer of transuranic (TRU) waste to the WIPP site unless a bypass from Los Alamos were built because of his concern about transporting the waste down the main thoroughfare of Santa Fe.

New Mexico lawmakers, although strongly opposed to WIPP, are resigned to the fact that the facility will eventually open—they "just want compensation for their state" in the form of payments equivalent to taxes in return for housing WIPP. In line with these concerns, the subcommittee drafted a substitute bill that included $397 million payment. Kostmayer said, however, that he did not believe that the money appropriated had been properly justified. "New Mexico is entitled to some compensation for taking WIPP," Kostmayer said, "but not a complete raid on the federal treasury." He predicted that when the bill goes to the Senate an effort will be made to raise the amount of money to compensate New Mexico. "I will resist the effort as best I can in conference," Kostmayer asserted. Richardson, who has been a major voice of opposition to WIPP, introduced five amendments that would, in his words, ensure that more reasonable safety measures would be included in the bill. His five amendments would:

1. Specify that WIPP would hold only 5.6×10^6 ft^3 of transuranic waste instead of 6.2×10^6 ft^3 and issue regulations within six months of test phase instead of one year.

2. Prohibit the transportation of transuranic radioactive waste to or from WIPP except in packages certified for such transportation by the NRC and have satisfied NRC's quality assurance provisions.

3. Prohibit transporting waste from the Los Alamos National Laboratory to WIPP until all the funds necessary for the cost of construction of the Santa Fe bypass have been appropriated by Congress or the State of New Mexico, or the Santa Fe bypass has been completed.

4. Appropriate funds to perform annual reviews of activities at WIPP.

5. Subject each test plan to review and approval by the state of New Mexico.

About a week after the Subcommittee on Energy and the Environment marked up the land-transfer bill, members of the Armed Services Committee heard various legislative proposals regarding this issue.[4] Congressional views on the land transfer are divided into two groups: One, led by Rep. Kostmayer, would require newly revised EPA regulations regarding disposal of TRU radioactive waste to be in effect *before* WIPP receives its first barrel of waste for the test phase. The second group, led by Reps. J. Skeen (R-N. Mex.) and R. H. Stallings (D-Idaho), argues that, if revised EPA regulations must be in place before WIPP can begin testing, the facility could be delayed another four to five years. DOE endorses the Bush Administration's bill, which is similar to the Skeen–Stallings measure, and views the Kostmayer bill as detrimental to the goal of opening WIPP on schedule.

According to L. P. Duffy, DOE's Director of Environmental Restoration and Waste Management, the Kostmayer bill would seriously affect the WIPP test phase and performance assessment program. Duffy felt that "the Department will be ready to receive the first transuranic waste shipments for testing this July [1991]. There is consensus within the scientific and technical community that a sufficiently firm basis exists to initiate the WIPP test phase so that performance assessment data can be collected, analyzed and integrated into a compliance determination."

The EPA testimony at the hearing stated that "as far as the present authorities of the EPA are concerned the DOE has met the requirements that are necessary for it to embark on the test phase activities."

Other states that are temporarily holding TRU waste pointed out that it might be more hazardous to continue to store this waste at the various sites than to begin storing it in test rooms at WIPP. Idaho's governor declared a ban on receiving any more TRU waste within the state, pending the initiation of tests at WIPP. Rep. W. Richardson (D-N. Mex.), however, insisted that WIPP should be required to comply with all EPA standards before any waste could be emplaced. Pointing to past weaknesses in other defense waste facilities, Richardson felt that, in the wake of recent reports about the instability of the rooms prepared for testing inside WIPP, DOE should take great care before shipping waste to WIPP. The concern about the soundness of the excavated rooms surfaced in testimony before another House committee in June 1991 (Ref. 5), where it was learned that the complex had severe structural problems, had experienced rock falls, and may not be structurally sound enough for DOE to conduct its proposed five-year test program. Richardson stated that he was committed to opening a safe WIPP and that he planned to introduce several amendments to the land withdrawal bill. He argued that adequate funds must be appropriated to ensure safe transportation routes and emergency response preparedness.

Duffy discussed the nature of the tests that are important for compliance determination and would be conducted during the test phase. The planned tests are to include some experiments that use radioactive materials. Non-radioactive tests include large-scale seal tests, ongoing rock mechanics tests, and several hydrologic tests, Duffy said. Laboratory tests to establish solubility and limited scoping data on gas generation rates to help guide the balance of the test program and to be used in the WIPP performance assessment are also going on in parallel, Duffy noted.

Chairman J. Spratt (D-S.C.) questioned the necessity of performing any tests at the WIPP site. He cited critics' complaints that DOE was trying to "get to WIPP because possession is nine-tenths of the law" and that most of the needed tests could be performed in laboratories. Duffy explained that it is important for DOE to demonstrate its ability to dispose of radioactive waste at the actual site safely. "We are not going to know for certain if it is safe until we demonstrate our capability at WIPP," Duffy asserted.

ACTIVITIES TO DESIGNATE YUCCA MOUNTAIN AS AN HLW REPOSITORY CONTINUE

The 1982 Nuclear Waste Policy Act, along with its amendments, provides the framework for the nation's program for the disposal of civilian high-level radioactive waste (HLW). It charges DOE to manage the permanent disposal of HLW. The 1987 amendments to the Act designated the Yucca Mountain site in Nevada to be characterized for suitability as a mined geologic repository. The amendments also created the Nuclear Waste Technical Review Board (NWTRB) to evaluate the scientific and technical validity of DOE's activities in this regard as well as its activities related to packaging and transporting high-level waste.

Many experts believe that the resolution of the problem of storage of high-level radioactive waste is one of the key problems that must be solved to revitalize nuclear energy in the United States. So far little progress has been made in this area. DOE believes that the problem "is a political one and not a technical one." We shall present below a summary of the ongoing struggle over the proposed HLW repository at Yucca Mountain.

Federal Government Tries to Bypass Nevada in Its Effort to Obtain Permits for Characterization of Yucca Mountain

The DOE and the Administration were trying to pass legislation to break the deadlock with the state of Nevada over the Yucca Mountain HLW site characterization. To this end, S. 570, known as the "National Energy Strategy Act," was introduced in the Senate[6] to amend the Nuclear Waste Policy Act to exempt all site characterization activities at Yucca Mountain from state, local, or tribal regulatory authority. The bill would also de-link the development of a Monitored Retrievable Storage (MRS) facility from a requirement to meet certain milestones in the development of a permanent repository.

A Senate Energy and Natural Resources Committee hearing in mid-May 1991 (Ref. 7) and an NWTRB report to Congress in late June 1991 (Ref. 8) dealt with the question of whether the Federal Government should take over responsibility for site characterization activities at Yucca Mountain. DOE testimony asserted that the state of Nevada had intentionally delayed review of the first of the 15 to 20 permit applications that DOE requires for site characterization and exploration at Yucca Mountain. Nevada, however, claimed that it was treating DOE "like any other permit applicant," which, said the State, is what DOE requested. Nevada claimed that any legislation that would strip it of its right to enforce federal health and environmental laws would set a precedent by which any state could be legally prohibited from protecting its citizens.

J. Bartlett, Director of DOE's Office of Civilian Radioactive Waste Management (OCRWM), sees the legislation as the only possible way to overcome delays in processing DOE permit applications. As proof of Nevada's intent to slow progress at Yucca Mountain, Bartlett cites that the state has processed eight other permit applications for surface mining since DOE applied for its permits, which remained unprocessed by the middle of 1991.

Deputy Secretary Moore assured the Committee that if S. 570 passed "DOE's obligation to follow environmental and other applicable legal requirements would not be removed," but instead, "monitoring authority would shift from the state to appropriate agencies within the executive branch of the federal government." He also noted that the bill allows the state to regain its authority if it becomes "willing to cooperate in good faith and enter into an agreement with . . . [DOE] governing site characterization."

The EPA endorsed the bill; under the provisions of S. 570, EPA would become responsible for characterizing the site and enforcing environmental regulations. The EPA testimony asserted that the Agency was ready and able to assume the characterization activities and supported provisions of the bill which ensure that the site will be characterized in a timely manner. The EPA believes that the bill is "strictly limited to the work necessary to characterize a potential site" and "does not extend to the determination of whether to site a repository at Yucca Mountain."

The state of Nevada, through its Agency for Nuclear Projects (NANP), defended itself against accusations of foot-dragging on DOE's permit applications. It pointed to a General Accounting Office (GAO) report published in April 1991, which states that DOE was not ready to begin characterization activities until February 1991. Thus, said Nevada, DOE itself was to blame for any delay in characterization activities. R. Loux, executive director of NANP, stated that the proposed law "provides the DOE with a return in part to self regulation."

Legal counsel for the Natural Resources Defense Council (NRDC) argued that the Federal Government already had constitutional authority to preempt the state in this matter if it was in the "paramount national interest." Therefore, said NRDC, S. 570 is not required. NRDC also said, however, that if the administration did decide to preempt the state, it would "set a very scary precedent" for loss of state's rights to the Federal Government.

The National Association of Regulatory Utility Commissioners (NARUC) pointed out that the public utilities industry has a vested economic and safety interest in the creation of a repository. The industry has paid millions into a waste fund, and some plants are nearing the end of their storage capacity for waste, which creates a potential safety hazard in the future. "You have our dollars, and we have your waste," said NARUC Vice Chairman C. Robinson, who urged the subcommittee to act immediately to resolve the waste problem.

Dr. D. U. Deere, chairman of the NWTRB, was asked two questions by the Committee: (1) Is DOE prepared to initiate site-characterization activities at Yucca Mountain? and (2) is there any reason to disqualify the Yucca Mountain site at this time? Deere answered the first question by saying that in NWTRB's view DOE was fully prepared to begin site-characterization activities as soon as it had gained access to the site. In response to the second query, Deere said that there appeared to be no scientific or technical reasons to abandon the site at this time.

Two of Nevada's Congressional officials, Sen. R. H. Bryan and Rep. J. H. Bilbray, voiced scathing criticism of S. 570. They argued that the bill would set a precedent by which a state could be legally prohibited from protecting its citizens. Further, Bryan stated, the people of Nevada could not be expected to put their trust in DOE, claiming that DOE's "track record of ineptitude is unparalleled." He added that Nevada had vowed to continue the fight to keep a waste dump for radioactive materials out of the state. "This is just the beginning of what promises to be a long struggle to resist this legislation. The health and safety of Nevadans cannot be compromised."

Meanwhile, the Senate Energy Committee in mid-June 1991 approved a bill (S. 1138) that would enable the Federal Government to designate Yucca Mountain as a nuclear waste repository.[9] The measure, passed by the committee, would allow DOE to begin construction of an MRS facility at Yucca Mountain. Current law calls for a permanent facility to be built before a temporary one could be constructed.

Volcanic Hazards May Decide Fate of Yucca Mountain

Although the political debate continued, the technical issue of whether Yucca Mountain is susceptible to volcanic activity within the next 10 000 years may be decisive for the future of the site. In March 1991 the NWTRB's Panel on Structural Geology and Geo-engineering convened a meeting to discuss recent studies relating to volcanic hazards in the vicinity of the Yucca Mountain site.[10] Representatives from DOE, NRC, the Electric Power Research Institute (EPRI), and the state of Nevada attended to discuss various aspects of this topic. According to D. C. Dobson, the acting director of the regulatory and site evaluation division at DOE, DOE believes that "a performance-based probabilistic approach is appropriate for evaluating volcanic hazards" at the proposed repository site. Further, Dobson said he believed that the current estimates and data are reasonable and conservative, geologically speaking. DOE fully believes that an evaluation of the possibility of volcanic activity at Yucca is feasible, Dobson asserted. "We think we can bound the probabilities at Yucca Mountain from a variety of different perspectives," said Dobson, specifically mentioning cone and volcanic rate estimates.

C. Johnson, administrator of technical programs for the state of Nevada, contended that in the past DOE had "failed to adequately address the issue" and that their "approach to resolution of the issue led to a false conclusion" that there was no real threat of volcanic activity in the area. But Johnson did give credit to DOE for redoubling their efforts and for renewed interest in the issue of volcanism in the region.

Two European Countries Have Lessons to Teach the United States

According to the NWTRB, Sweden and Germany are making good progress in developing high-level radioactive waste repositories, which is due in part to greater flexibility in governmental regulatory criteria.[11]

In the spring of 1990, the members of the Board traveled to Germany and Sweden to assess the progress being made in developing programs for managing high-level radioactive waste. They were specifically interested in collecting information on technologies and policies that could be applied to the U.S. program. They concluded that the Swedish and German programs were "well conceived and making progress" with underground research under way and interim storage as an important part of the waste disposal strategy in both countries.

The regulatory criteria used in Germany and Sweden to design and build a repository are based on radiation dose limits to individuals, in contrast to the criteria used in the United States, which use specific containment standards criteria for regulation. The NWTRB agreed that the Swedish and German systems seem to provide more flexibility to develop the best possible repository design.

On licensee applications, the Board noted that Sweden and Germany make less of a distinction than the United States between the applicant for a repository license and the licensing agency. Although the Board conceded that the American system ensures a more independent review for a repository, the U.S. arrangement has a tendency to create adversarial interagency relationships.

Although admitting that some research results are being shared by the three countries, the Board noted that the need exists for more information sharing on the use of engineered barriers, container design and development, thermal loading and waste aging, grouting and back-filling techniques, use of mechanical versus drill-and-blast tunnel-boring methods, and assessment methodologies for long-term repository performance.

THE LINKAGE BETWEEN THE MRS SCHEDULE AND A PERMANENT HLW REPOSITORY

The permanent HLW repository being characterized at Yucca Mountain and the MRS program being developed by DOE are separate but related projects. The question of exactly how the two projects should be linked was the topic of a Senate Subcommittee on Nuclear Regulation hearing toward the end of May 1991 (Ref. 12). At issue was the concern that acceptance of an MRS might result in a de facto permanent repository if difficulties in siting a permanent repository cannot be resolved. Under current law, the schedule for siting and building an MRS is linked to the ultimate siting and construction of a permanent repository. Since progress on a permanent repository continued to look hopelessly bogged down by the difficulties in passing legislation and by endless litigation, section 512 of S. 570 was designed to allow construction of an MRS to proceed independently of the status of an HLW repository, thus severing the linkage.

Nuclear Waste Negotiator D. Leroy, the former chairman of the MRS Commission, sees a change in the linkage stipulation as probable.[13] His office is charged with finding an MRS site and negotiating the terms of its acceptance. To Leroy "it appears that the construction of an MRS will precede the construction and completion of a repository. Any reasonable agreement that is presented to Congress by the Negotiator for an MRS is likely to recommend that the existing linkages be modified in order to proceed to construct the MRS." Leroy emphasized that some form of assurances would nevertheless be necessary before a host would consider construction of a facility.

The crux of the argument against enacting Section 512 of S. 570 is that any MRS host state would be extremely loath to accept an MRS without statutory assurance that the waste would go, at some definite point in time, to a permanent repository. This position was presented by the NRDC (Ref. 13), who also said that it makes "basic public policy sense" not to site an MRS unless the public gives its consent.

Meanwhile, DOE's OCRWM, which is currently involved in the siting and construction of an MRS, hopes to have a facility prepared to accept limited waste as early as 1998 (Ref. 14). The OCRWM has given official notice of its intent to issue a restricted eligibility solicitation inviting the submission (by eligible entities) of applications for grants of financial assistance. These grants would be for studies to assess siting feasibility at a location under the jurisdiction of one of the eligible groups.

NRC TO REASSERT REGULATORY AUTHORITY IN IDAHO

In June 1991 the NRC reasserted its regulatory authority over the possession and use of by-product, source, and special nuclear material in the state of Idaho.[15] Governor Cecil Andrus, in a Mar. 25, 1991, letter to the Commission, advised that Idaho, which had been an Agreement State since 1968, could no longer carry out its responsibilities as an Agreement State because of severe budget constraints and other compelling reasons. The action, effective Apr. 26, 1991, was taken to ensure that the public health and safety would be protected.

Licensees in Idaho (there are about 130 of them) were advised of the Commission's action in an Order dated Apr. 11, 1991, and the NRC staff worked with Idaho authorities to ensure an orderly transition in regulatory authority. As part of this effort, the NRC Staff was planning to hold public workshops in the State to explain NRC rules, fee schedules, and enforcement policies. In the meantime, the Idaho licenses would remain in effect until they could be revised, if necessary, to meet NRC requirements.

FEDERAL DISTRICT COURT RULES THAT STATES WITH LLW REPOSITORIES MUST CONTINUE ACCEPTING MICHIGAN'S WASTE

In a decision issued on June 18, 1991, the United States District Court for the Western District of Michigan, Southern Division, ruled that Nevada, South Carolina, and Washington would be required to accept low-level radioactive waste (LLW) generated in Michigan until Dec. 31, 1992 (Ref. 16). In addition, the Court enjoined the state officials from denying Michigan waste generators access to disposal facilities within their states prior to Jan. 1, 1993.

The Michigan Coalition of Radioactive Materials Users (MICHRAD), an association representing Michigan generators (including hospitals, laboratories, utilities, and industrial users), brought the suit when officials of Nevada, South Carolina, and Washington cut off Michigan generators' access to the only licensed disposal facilities in the country. The three states asserted that Michigan had made insufficient progress toward developing its own regional disposal facility and as a result was no longer in compliance with a milestone in the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA), upon which the right of access was based.

Congress, with input from the National Governors Association, passed the 1985 Act to ensure the development of additional low-level waste sites by Dec. 31, 1992. The country was divided into regional compacts in which one state would be designated to receive that region's waste. The 1985 Act provided various incentives and penalties to encourage states without sites and compact states to develop disposal capacity by Dec. 31, 1992. The Act guaranteed that, during the development of such sites, the states without sites could continue to dispose of a certain amount of low-level waste in the states with sites in the transition period from Jan. 1, 1986, to Dec. 31, 1992.

The 1985 Act, however, allowed states with sites to deny access to their disposal sites if states without sites did not meet certain milestones outlined in the LLRWPAA. One of the milestones was a 1988 deadline for submission by each regional compact of a plan to site a disposal facility in a host state. In 1988, Michigan submitted such a plan for the Midwest Compact but had difficulty adhering to its own proposed schedule for subsequent activity.

The Court determined that "sited states are only authorized to deny access if a milestone is not reached. Under the plain language of the Act, Congress has chosen dates certain and identified precisely what is required to be done by those dates. Once the 1988 milestone is reached, the sited states do not have authority to monitor a state's progress under the siting plan. There is no provision in the Statute that even hints of authorizing the sited states to monitor the good faith of the non-sited states in carrying out their site plan or abiding by the proposed schedule for development of a disposal facility."

According to C. J. Cooper of the Washington D.C. law firm of Shaw, Pittman, Potts & Trowbridge, co-counsel for MICHRAD, "The Court's decision found that Michigan, which has been selected as the host state for the Midwest Compact, has met the Act's milestones. The defendants' contention that Michigan had somehow not continued to make progress after meeting the milestones is not a valid argument for denying access to the waste sites."[16]

Cooper said the decision has important ramifications for other states that are having difficulty developing their own disposal facilities. "If a state submitted a siting plan in 1988 and certified in 1990 that it would be responsible for managing its own waste after 1992, it should be entitled to continue to use the existing facilities in the three states through 1992."

ACNW COMMENTS ON SEVERAL ISSUES

The Advisory Committee on Nuclear Waste (ACNW) sent six letter reports to the NRC during April, May, and June 1991. Four of these letter reports will be briefly discussed and excerpted here. The other two letter reports concern upcoming activities of the ACNW (Ref. 17) and its plans for reviewing Regulatory Guides on revised 10 CFR Part 20 (Ref. 18).

ACNW Comments on Draft SECY Paper on Dealing With Uncertainties

At its meeting at the end of April 1991, the ACNW reviewed a copy of the draft SECY paper, "Staff's Approach for Dealing with Uncertainties in Implementing the EPA High-Level Waste Standards"[19] The NRC staff wrote this paper in response to a request from the Commission for an explanation of the management of uncertainties during the process of evaluating compliance of a proposed repository with the probabilistic standards of the U.S. Environmental Protection Agency (EPA). Comments and concerns expressed in this meeting were transmitted to R. M. Bernero, Director of the Office of Nuclear Material Safety and Safeguards. Excerpts from the letter are as follows.

> The draft SECY paper and its accompanying document provide a broad view of the uncertainties that will need to be addressed during site characterization and the subsequent licensing process Although the draft SECY paper includes discussion of methods to reduce uncertainties, we believe the staff has insufficiently clarified its role in the management of uncertainties that will remain after a license application is submitted The draft SECY paper is also substantially silent on (1) the general program plan envisioned by the NRC staff for managing uncertainties, (2) the way in which rulemaking and similar protocols will be used to manage uncertainties that are likely to become important at the time of license hearings, and (3) the distinction between the role of the NRC and that of the U S Department of Energy in reducing and managing technical uncertainties At the same time, the draft SECY paper includes extensive coverage of topics that could be interpreted as not being pertinent to the questions that need to be addressed One example is the discussion of the benefits to be derived from the existing version of the EPA Standards The discussion of collective versus individual dose limits should also be removed from the SECY paper
>
> Although the draft paper is partially responsive to the request of the Commission for a discussion of the management of uncertainties, there is a need to develop a program plan that (1) establishes guidelines for developing responses to a broad range of uncertainty issues, (2) describes the bases for actions by the staff, for example, the method of balancing reliability and risk, and (3) serves as a guide to the preparation of additional reports that systematically explore the application of the overall plan to various parts of the licensing process, such as the approach to reconciling expert judgments that conflict Such a plan would provide assurance of long-term regulatory consistency and completeness, in essence, it would serve as a "road map" The existing draft paper and our discussions with the NRC staff can readily serve as a beginning for the preparation of a program plan
>
> We believe that the staff is approaching the difficult and complex topic of uncertainty issues with growing insight Although the present draft SECY paper represents an improvement over the earlier version, it demonstrates the need to organize the variety of issues to be addressed so that uncertainties are minimized and managed satisfactorily, leading to the formulation of defensible policies Some parts of the draft paper, particularly portions of section 2 and much of section 3, could, after revision, be issued as a partial response to the Commission's request

ACNW Comments on Dose Limits and Radionuclide Release Limits

The ACNW has been developing comments, thoughts, and suggestions relative to individual and collective dose limits and radionuclide release limits Because the NRC's Office of Nuclear Material Safety and Safeguards was reviewing these same topics, the ACNW provided it with comments, which are given as follows[20]

> <u>Basic Definitions</u>
>
> As a basic philosophy, individual dose limits are used to place restrictions on the risk to individual members of the public due to operations at a nuclear facility If the limits have been properly established and compliance is observed, a regulatory agency can be confident that the associated risk to individual members of the public is acceptable Because the determination of the dose to individual members of the public is difficult, the International Commission on Radiological Protection (ICRP) has developed the concept of the "critical group" and recommends that it be used in assessing doses resulting from environmental releases As defined by the ICRP, a critical group is a relatively homogeneous group of people whose location and living habits are such that they receive the highest doses as a result of radionuclide releases The group may be real (in which case their actual habits may be known or predicted) or hypothetical (in which case their habits may be assumed, based on observations of similar groups)
>
> The dose to individuals within the critical group is assumed to be that received by a typical member of the group The purpose of this approach is to ensure that members of the public do not receive unacceptable exposures while, at the same time, ensuring that decisions on the acceptability of a practice are not prejudiced by a very small number of individuals with unusual habits If the

number of people being exposed is large, the question often arises as to how to quantify the societal impact of the individual exposures The collective dose concept was developed for expressing that impact in a quantitative manner and, as such, it is a numerical expression of the summed doses to a given population

In many respects, placing limits on total radionuclide releases from a nuclear facility is comparable to placing a limit on its total societal impact In other words, placing a limit on the quantity of a given radionuclide that can be released is equivalent to placing a limit on the total societal impact that the facility can exert This was the basis used by the U S Environmental Protection Agency (EPA) in setting release limits for a high-level radioactive waste repository, and it relates directly to EPA's basic criterion that the number of health effects should not exceed 1,000 during the first 10,000 years

Underlying Assumptions

Although it is generally accepted that the dose received by an individual is a reasonable expression of the associated risk, it is questionable whether the collective dose is a true measure of the societal impact of the aggregate of exposures to individual members of a population Implicit in the concept of collective dose is the assumption that the linear hypothesis is correct, that is, that there is a linear (non-threshold) relationship between the total dose to a population group and the associated health impacts

In many ways, application of the collective dose concept leads to a paradox At high doses and high dose rates where the risk coefficients are best known, the concept of collective dose cannot be applied since the dose–response curve is nonlinear, at low doses and low dose rates where linearity between dose and the associated health effects is assumed to apply, the risk coefficients are far less certain This leads to additional restrictions in the application of the collective dose concept, as follows

- The exposed population must be well known with respect to size and possibly age, sex, and temporal distributions
- The exposure pathways must be characterized for the population at risk
- Individual contributions to the collective dose must consist only of doses to the whole body, or to specific organs or tissues for which stochastic risk coefficients are known

In short, application of the collective dose concept requires detailed knowledge of the exposed population and the radiation doses to its members The collective dose concept is valid for representing the collective risk only if both of these factors can be described and quantified, and it should be used for risk assessments only if the associated uncertainties are sufficiently small that the calculated collective dose itself is within an acceptable range of uncertainty In addition, it is important to note that a high individual risk to a small number of people is not necessarily the same as a low individual risk to a large number of people, even though the collective dose may be the same For this reason, expressions of societal risk in terms of collective dose should always include detailed data not only on the number of people exposed, but also on the number of people receiving exposures within each dose range Although collective dose can be used as a surrogate for societal risk, its interpretation requires care

Truncation of Collective Dose Calculations

On a theoretical basis, there is no justification for excluding the application of the linear hypothesis to the evaluation and interpretation of the societal impact of low doses and low dose rates on population groups This hypothesis, in fact, has been generally accepted by the scientific community, including organizations such as the National Council on Radiation Protection and Measurements (NCRP) and the ICRP, as a valid basis for estimating the stochastic risks associated with low doses of ionizing radiation If one accepts this observation, calculations of collective doses should include the doses to all individuals within the population group, regardless of how small the associated doses and/or dose rates may be At the same time, however, it is important to recognize that there may be cogent reasons for not including within collective dose calculations extremely low doses to individual members of a population group Several approaches that have been proposed and/or applied to justify such omissions are discussed below

Following the concept that certain risks to individual members of the population are negligible, the NCRP has recommended (under what it defines as the concept of a "Negligible Individual Risk Limit") that annual doses to individual members of the population that are less than 0 01 mSv (1 mrem) be excluded from collective dose calculations In interpreting this recommendation, however, it is important to understand the underlying principle on which it was based Informal discussions with representatives of the NCRP revealed that truncation in this case was considered to be acceptable from the standpoint of societal impact, because the burden on society represented by any additional cancers among people receiving exposures in this dose rate range would not necessitate any additional medical facilities Another approach for truncation that has been informally suggested by representatives of the NCRP is that it might be permissible to discard a collective dose (calculated on the basis of extremely low dose rates to members of an exposed population) provided that the associated collective dose would not be estimated to result in one additional cancer

Variations in the dose rates from natural background radiation sources have been proposed as another basis on which to truncate collective dose calculations The contribution to collective dose from natural sources is large relative to that from many artificial sources Consequently, it is often difficult to measure in a meaningfully quantitative manner very low dose rates to individual members of the population that arise from artificial sources Thus, although there may be no biological basis for excluding very low dose rates from collective dose calculations, there is justification for excluding them on a statistical basis because of the uncertainties in the associated calculations

Determinations of Compliance With Standards

From the previous discussion, it follows that the establishment of limits on the concentration of individual radionuclides in various environmental media (e g , air and water) is comparable to the establishment of dose limits for individual members of the population Likewise, the placement of limits on total radionuclide releases from a nuclear facility is comparable to the establishment of limits on the associated permissible collective doses to the affected population In terms of the determination of compliance with a set of standards, it is readily possible to measure the concentrations of individual radionuclides in various environmental media, and it is similarly possible to estimate the associated doses to individual members of the population In contrast, estimates of the total releases of radionuclides from a nuclear facility would require not only knowledge of the concentrations of individual radionuclides in all environmental media, but also the determination of the rate of movement (transport) of each radionuclide (including the evaluation of site-specific pathways) within all such media from the facility to the accessible environment Similar uncertainties would accompany estimates of the associated collective doses

Summary

In summary, the Committee offers the following statements on the benefits of the application of various limits for determining the public health risks associated with nuclear operations

1 Individual dose limits can be used to limit the risks to individual members of a population group

2 Collective dose limits can be used to limit the societal impacts of doses to a large number of individuals The accuracy of collective dose as a measure of societal risk, however, depends on the validity of the linear (non-threshold) hypothesis in assessing the stochastic effects of ionizing radiation

3 Collective dose calculations are representative of societal risk only if certain conditions are satisfied, namely, the exposed population is defined and characterized with respect to size, age, and sex, the distribution of doses to individual members of the population is within a limited range, the exposure pathways have been characterized for the population at risk, and individual contributions to the collective dose consist only of doses to the whole body, or to specific organs or tissues for which stochastic risk coefficients have been adopted

4 Techniques for measuring the concentrations of individual radionuclides in various environmental media, and for estimating the associated dose rates to individual members of the population, are readily available, and compliance with such limits can be determined In contrast, the measurements that would be required to determine the total releases of individual radionuclides from a nuclear facility and estimations of the associated collective dose to all offsite population groups would be difficult

5 Given the general acceptance of the linear hypothesis, there is no biological basis on which to truncate calculations of collective doses Nonetheless, regulators must recognize that estimates of dose rates from artificial radiation sources, that represent only a few percent of those from natural radiation sources, carry with them large uncertainties and relatively little aggregate risk Such uncertainties may well serve as a basis for truncating collective dose calculations at very low dose rates without adverse impacts on estimates of the associated risks

ACNW Comments on EPA's "Three Bucket Approach"

With the issuance of Working Draft 3 of 40 CFR Part 191, the proposed revised standards for the management and disposal of high-level radioactive wastes, EPA requested comments on its proposed "three-bucket approach" for classifying events that may affect repository performance. The three buckets are based on an assumed life of the repository of 10^4 years.

Bucket Number 1 Scenarios with cumulative intrusion frequencies greater than 1/10: quantitative probabilistic performance assessment.

Bucket Number 2 Scenarios with cumulative intrusion frequencies between 1/10 and 1/10,000: individual scenario, deterministic analysis, comparison to 10 × the release limits.

Bucket Number 3 Scenarios with cumulative intrusion frequencies less than 1/10,000: analysis not required.

At its meeting at the end of April 1991, the ACNW provided the following comments:[21]

> In general, we endorse the three-fold classification system outlined in the enclosure, [*Editor's note the enclosure referred to defines the "three buckets" described above*] and we believe it will be helpful in addressing the problems of assessing inadvertent human intrusion We also endorse the deterministic treatment of scenarios that are assigned to "bucket number two "
>
> We accept the fact that the presence of natural resources represents a potentially adverse condition [10 CFR 60 122 (c)] If there are potential resources present at a site in large enough amounts to create a high probability for human intrusion, the site should be rejected We expect that no scenario involving inadvertent human intrusion will be assigned to "bucket number one "
>
> As part of our continuing study of the "three-bucket approach," we are evaluating the bounding probability limit

for distinguishing between scenarios that are unlikely ("bucket number two") and very unlikely ("bucket number three").

ACNW Comments on Human Intrusion Concerns in the Licensing of an HLW Repository

In May 1990 the NRC recommended that EPA standards for the disposal of high-level radioactive waste be revised to permit the application of a separate approach for evaluating the potential impacts of human intrusion. According to D. A. Moeller, Chairman of the ACNW, one approach for evaluating human intrusion in the case of the geologic repository would be to apply techniques similar to those used by the NRC in assessing the threat of sabotage at nuclear power plants.[22] In evaluating this threat, the NRC uses a deterministic rather than a quantitative probabilistic approach. The NRC approach recognizes the inherent uncertainties associated with the application of quantitative probabilistic techniques in assessing an issue of this nature. Therefore, by letter report, Moeller provided the NRC with a paper that summarizes the NRC approach in the treatment and evaluation of the sabotage threat at nuclear power plants and addresses the issue of human intrusion as treated in the EPA standards. The introduction and conclusions from this letter report, prepared by S. E. Mays (ACRS/ACNW Fellow) at the request of the ACNW, is provided below. [Emphasis on specific words and sentences is Mays'.]

INTRODUCTION

Nuclear power plants have engineered features and proposed HLW repositories have engineered and geologic features that serve to limit the likelihood of release of radioactive material to the environment. In the case of nuclear power plants, several engineered barriers exist including the fuel cladding, the reactor coolant system boundary, and the containment. For spent fuel at a nuclear plant, the barriers include the fuel cladding and the spent fuel pool (or dry cask storage at some locations). For a HLW repository the proposed barriers include the fuel cladding, the containers for the spent fuel, and the geological formation (analogous to the reactor containment).

Human actions such as sabotage or human intrusion have the potential to bypass the features that limit the likelihood of release of radioactive material to the environment. *While the intent of the participants and the nature of these two actions are different, such events are difficult to analyze by probabilistic techniques and at least sabotage is not so treated. This paper examines the extent that the two agencies use probabilistic techniques to regulate protection from these acts of commission. It is not intended to equate the physical acts themselves nor to state that the approach suggested here is the final word on the subject.*

The NRC and EPA have regulations requiring licensees to demonstrate their ability to maintain the integrity of the features against certain acts of commission. In the case of nuclear power plants, physical security requirements for protection against sabotage are contained in 10 CFR 73.55. For a HLW repository, the EPA requirements for human intrusion (HI) are contained in an appendix to 40 CFR 191.

While both agencies recognize the potential for human actions to bypass these protective features, the use of probabilistic techniques in the licensing and regulatory process is vastly different. Briefly stated, the EPA regulations require a quantitative probabilistic analysis (called a performance assessment) of the performance of the protective features of a repository over a 10,000 year period. This assessment must include human intrusion scenarios explicitly. The NRC approach with respect to sabotage at nuclear power plants, on the other hand, eschews quantitative probabilistic criteria in favor of a deterministic evaluation supported by qualitative use of probabilistic analyses.

The purpose of this paper is to compare the methods used by the NRC and EPA to regulate protection from sabotage at reactors and inadvertent human intrusion at a potential HLW repository. The paper specifically addresses the use of (or the lack of) probabilistic techniques in their regulations and applications. While there may be concerns regarding the similarity of the events themselves (and therefore the applicability of comparing the types of regulation) and whether either agency has come upon the ultimate methodology for regulating them, this paper compares the regulations and applications *as they currently exist.* It is for the reader to determine the applicability of these techniques to the regulation of protection against sabotage at reactors and human intrusion at a HLW repository.

CONCLUSIONS

Human actions have the potential to adversely affect the protective systems that limit the release of radioactive material to the environment. The NRC and EPA have chosen vastly different ways to deal with such actions in their regulations for nuclear power plants and HLW repositories.

The EPA has opted for a quantitative, probabilistic analysis that includes human intrusion as one of its parts. The EPA guidelines specify the frequency of the HI events and the effectiveness of controls to prevent intrusion for the analysis. No such specification of frequencies or effectiveness of engineered systems for other scenarios is stipulated.

The NRC has opted for a deterministic approach for plant security. In a method similar to their treatment of design basis events, the NRC has specified a threat level that security plans must account for. The NRC requires identification of vital equipment and the areas encompassing vital equipment. Probabilistic techniques are used internally by the NRC staff to produce qualitative results that support the evaluation of the effectiveness of licensee security programs. Quantitative risk curves are not a licensing requirement for this issue. In fact, the NRC does not have any licensing criteria that require a risk curve comparison to a numerical standard.

This paper addresses the methods that the NRC and the EPA use to regulate protection from sabotage at reactors and inadvertent human intrusion at a potential HLW repository. While there may be concerns regarding the similarity of the events themselves (and therefore the applicability of comparing the types of regulation) and whether either agency has come upon the ultimate methodology for regulating them, this paper compares the regulations and applications *as they currently exist.* It is for the reader to determine the applicability of these techniques to the regulation of protection against sabotage at reactors and human intrusion at a HLW repository.

NRC CONSIDERS CONSENSUS-BUILDING STRATEGY TO GARNER SUPPORT FOR BRC POLICY

When NRC first published its Below Regulatory Concern (BRC) policy in the *Federal Register* in July 1990, the scope of the dissatisfaction it produced could not have been envisioned by the Commission. Public reaction was immediate and intense: one Senator noted with sarcasm that "what may be below the concern of NRC" is of tremendous concern to many others. The policy prompted introduction of legislation on the national, state, and local levels, with many public-interest groups and several industry organizations also criticizing the policy.

The BRC policy as described by NRC establishes a framework for making decisions on granting exemptions from Commission regulations in cases where radiation levels are "so low that they do not require the imposition of regulatory controls to ensure protection of public health and safety." With the introduction of this policy, NRC came under the unfriendly scrutiny of several lawmakers who pointed out that the policy permits radiation exposure levels higher than those endorsed by the EPA, the NCRP, and the International Atomic Energy Agency. Sen. G. J. Mitchell (D-Maine) also advertised the fact that BRC could permit up to 30% of the nation's low-level radioactive waste to be sent to ordinary landfills rather than to more protective low-level waste repositories.

Following the intense public reaction, NRC Chairman K. M. Carr suggested an evaluation of the use of a broad consensus-building process to develop a base of understanding and support for BRC. The Commission agreed. F. X. Cameron, Deputy Administrator for the Office of Licensing Support Systems at NRC, and H. Bellman, an independent arbitrator with experience in consensus building, began collecting information and performing their evaluation of a broad consensus-building process to develop a base of understanding and support for BRC. Cameron presented the findings to the Commission at the end of May 1991 (Ref. 23).

Cameron and Bellman began their assessment by interviewing 30 organizations–entities affected by BRC policy, including state governments, federal agencies, industry, and public-interest groups. Stating the need to complete the evaluation rapidly, Cameron noted that some groups conceivably affected by BRC were omitted from the interview process but added that input from these groups would be essential in any future consensus-building project. A sampling of some of the groups interviewed includes: officials from 12 states, representatives from regional low-level radioactive waste compacts, officials from DOE and EPA, and interest groups like the National Audubon Society, Sierra Club, and Natural Resources Defense Council. Cameron said repeated attempts were made to contact the interest groups Public Citizen, Greenpeace, and the Nuclear Information and Resource Service, but these groups refused to take part in any discussion of a consensus-building effort.

The groups who were contacted explored basic perspectives on BRC, possible participatory processes, substantive issues to be addressed and in what sequence this should be done, the general level of interest, other groups who should participate, and what might happen if a participative process were not implemented.

After reviewing the discussions, Cameron and Bellman concluded that a consensus process for the BRC policy is feasible. Cameron stated that "there seems to be a broad enough base of support for such a process among the groups that we interviewed" and noted that "the primary objective of the process would be the provision of advice by a consensus body to the Commission on the entire range of BRC issues." The outcome of the consensus body should be *advice* to the Commission, said Cameron. He pointed out that one of the biggest problems with the BRC policy is lack of communication among the affected groups.

"I do not believe that a negotiated agreement [rulemaking] is appropriate in this case," said Cameron, who reminded the Commissioners that public comment was tried previously and resulted in the current, much-maligned policy statement. Cameron said further that the BRC policy should be taken over by an external, paternal oversight group, which would sponsor the cause and lend credibility to any ultimate policy product. This core group would then sponsor the foundation of a larger "steering committee" where tentative goals on substantive issues would be formed and preliminary agendas–schedules set. The steering committee would also generate invitations for individuals to form a plenary body to adopt procedural ground rules and act as a clearinghouse for information sharing. Bellman and Cameron also suggested that the Commission should declare an immediate moratorium on implementation of the BRC policy to demonstrate NRC's commitment to encourage participation in the process. Both Bellman and Cameron, in answering questions from Commissioners at the hearing, emphasized that unanimity, or near unanimity of consensus is the most important goal. If it could be achieved, the source of the policy could not be seriously challenged, though the policy itself could be questioned

Nuclear Regulatory Commission Chairman Carr then asked Bellman and Cameron to specify the nature of the dissatisfaction of groups with current BRC policy. Cameron responded that the main problem was that public input was not sought until after the policy was formulated, and that, when comments were received, no response was given by the Commission. Commenters felt, said Cameron, that their comments or questions "fell into some black hole" at NRC. The Commissioners gave no immediate indication of their views on Cameron's suggestions. Meanwhile, congressional opposition to the BRC policy remained steadfast. On the same day as the NRC BRC hearing, Sen. G. J. Mitchell (D-Maine) introduced S. 1111, "a bill to protect the public from health risks from radiation exposure from low-level radioactive waste, and for other purposes", to the Committee on Environment and Public Works Mitchell's proposal would revoke the BRC policy and require NRC to give notice, comment, or an adjudicatory hearing before establishing a new low-level radioactive waste deregulation policy

Mitchell, along with Sen. W. S Cohen (R-Maine) and Sen. A. D'Amato (D-N.Y.), argued that NRC's BRC policy in effect prohibits states from enacting or enforcing restrictions on low-level waste disposal. All three Senators state that legislation is necessary to reaffirm the rights of states to be more protective of their citizens One provision of the bill provides federal facilities with the authority to pay fees assessed by a state for the search for and construction of a low-level waste site, a measure that Cohen said should help states in the low-level-waste siting process.

REFERENCES

1 *At Energy Clearing House* 37(14) 1 (Apr 5, 1991)
2 Activities Related to Waste and Spent Fuel Management, *Nucl Saf* 32(3) 402-415 (July–September 1991)
3 *At Energy Clearing House* 37(24) 6 (June 14, 1991)
4 *At Energy Clearing House* 37(25) 4 (June 21, 1991)
5 Hearing Before the House of Representatives Government Operations Committee's Subcommittee on Energy, Environment, and Natural Resources, June 1991
6 *At Energy Clearing House* 37(20) 6 (May 17, 1991)
7 *At Energy Clearing House* 37(20) 5 (May 17, 1991)
8 *At Energy Clearing House,* 37(25) 7 (June 21, 1991)
9 *At Energy Clearing House* 37(24) 1 (June 14, 1991)
10 *At Energy Clearing House* 37(18) 5 (May 3, 1991)
11 *At Energy Clearing House* 37(25) 7 (June 21, 1991)
12 *At Energy Clearing House,* 37(21) 3 (May 24, 1991)
13 *At Energy Clearing House* 37(21) 5 (May 24, 1991)
14 *At Energy Clearing House* 37(19) 5 (May 10, 1991)
15 NRC Press Release 91-39, Apr 12, 1991
16 *At Energy Clearing House* 37(26) 6 (June 28, 1991)
17 Letter to K M Carr, Chairman, U S Nuclear Regulatory Commission, from D A Moeller, Chairman, Advisory Committee on Nuclear Waste, *Program Plan for the Advisory Committee on Nuclear Waste,* cited in NRC Press Release 91 48, May 3, 1991
18 Letter to K C Rogers, U S Nuclear Regulatory Commission, from D A Moeller, Chairman, Advisory Committee on Nuclear Waste, *Review of Regulatory Guides on Revised 10 CFR Part 20,* cited in NRC Press Release 91-58, June 6, 1991
19 Letter to R M Bernero, Director, Office of Nuclear Material Safety and Safeguards, U S Nuclear Regulatory Commission, from D A Moeller, Chairman, Advisory Committee on Nuclear Waste, *Review of Draft SECY Paper Dealing with Uncertainties,* cited in NRC Press Release 91-58, June 6, 1991
20 Letter to R M Bernero, Director, Office of Nuclear Material Safety and Safeguards, U S Nuclear Regulatory Commission, from D A Moeller, Chairman, Advisory Committee on Nuclear Waste, *Individual and Collective Dose Limits and Radionuclide Release Limits* cited in NRC Press Release 91-48, May 3, 1991
21 Letter to K M Carr, Chairman, U S Nuclear Regulatory Commission, from D A Moeller, Chairman, Advisory Committee on Nuclear Waste, *Alternative Approach to the Probabilistic Section of the Containment Requirements in 40 CFR Part 191 (The Three Bucket Approach)* cited in NRC Press Release 91-58, June 6, 1991
22 Letter to K M Carr, Chairman, U S Nuclear Regulatory Commission, from D A Moeller, Chairman, Advisory Committee on Nuclear Waste, *Consideration of Human Intrusion in the Licensing of a High-Level Waste Repository* cited in NRC Press Release 91 48, May 3, 1991
23 *At Energy Clearing House* 37(21) 1 (May 24, 1991)

Operating Experiences

Edited by G. A. Murphy

Effects of Component Aging on the Westinghouse Control Rod Drive System

By K. Sullivan[a] and W. Gunther[a]

Abstract *An assessment of aging for the Westinghouse pressurized-water-reactor (PWR) control-rod-drive (CRD) system has been completed as part of the U S Nuclear Regulatory Commission Nuclear Plant Aging Research (NPAR) Program This study examined the design, construction, maintenance, and operation of the system to determine its potential for degradation as the plant ages This article presents selected results from this study*

The operating experience data were evaluated to identify the predominant failure modes, causes, and effects From our evaluation of the data, coupled with an assessment of the materials of construction and the operating environment, we conclude that the Westinghouse CRD system is subject to degradation that, if left unchecked, could affect its safety function as the plant ages

Ways to detect and mitigate the effects of aging are included in this article The current maintenance for the CRD system at 15 Westinghouse PWRs was obtained through a survey conducted in cooperation with the Electric Power Research Institute and the Nuclear Management and Resources Council The survey results indicate that some plants have modified the system, replaced components, or expanded preventive maintenance practices Several of these activities have effectively addressed the aging issue

In response to reactivity control signals that may be generated manually by the reactor operator or automatically by the reactor control system, electromechanical drive assemblies of the Westinghouse control-rod-drive (CRD) system physically position clusters of neutron-absorbing control rods within the core. By limiting any rapid changes in reactivity, such as those which may occur as a result of variations in plant load, the CRD system provides the principal means of ensuring that specified fuel design limits will not be exceeded. If plant operating safety limits are surpassed, the system permits control rods to fall by gravity into the core and thus cause a maximum negative reactivity insertion and result in a rapid shutdown of the reactor (scram).

By mitigating operational transients or accidents, this system performs a vital role in ensuring plant safety. Although failures within this system have never resulted in a total loss of reactivity control, the age-related degradation of its numerous components has initiated unplanned trips of the reactor (i.e., scrams), which often result in unnecessary challenges to the plant's safety equipment. The safety significance of this system has also been expressed in NUREG-1185, *Integrated Safety Assessment Report,* for one of the oldest Westinghouse pressurized-water reactors (PWRs). Following a discussion of operational problems experienced at this plant, the report concluded that "aging may be a factor in the perpetuation of control rod drive problems" and that failures of the CRDs "constitute a symptom of a plant problem which has safety significance." The need to address the time-dependent degradation of this system in the incipient stage is therefore a concern to both the industry and the U.S. Nuclear Regulatory Commission (NRC).

The Westinghouse CRD system was reviewed for the NRC's Nuclear Plant Aging Research (NPAR) Program to assess its potential for degradation as the plant ages.[1] This study examined the design, maintenance, and operation of the system and assessed the extent to which component aging, if left unchecked, could affect its operation.[2] As depicted in Fig. 1, the boundary of components considered includes the control rods, CRD mechanisms (CRDMs), power and logic cabinets, and associated in-

[a]Brookhaven National Laboratory, Department of Nuclear Energy, Upton, New York. Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

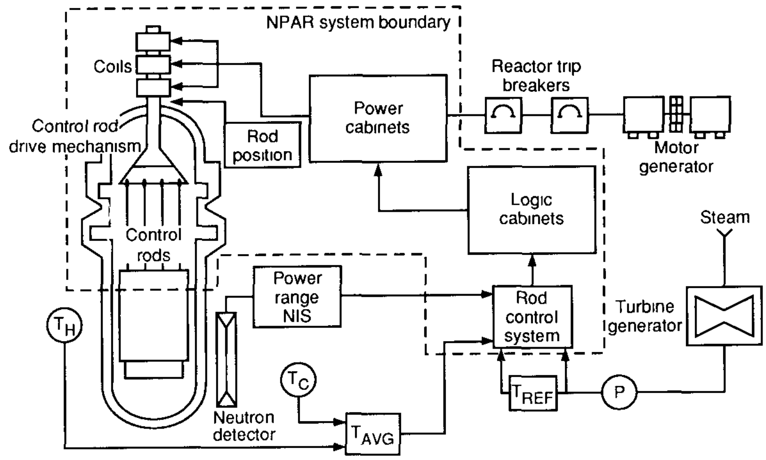

Fig. 1 Control-rod-drive system block diagram (Westinghouse pressurized-water reactors). NPAR, Nuclear Plant Aging Research Program; NIS, Nuclear Instrumentation System.

terconnecting cabling. Additionally, the rod position indication system (RPIS), which is actually an independent subsystem of electrical components that continuously senses and displays control-rod position information, was included in the scope of the study. In addition to the more obvious operational stressors this system is exposed to, such as cyclic wear and high ambient temperature, the study also considered the impact of operating demands and required testing on CRD system performance. This article presents a summary of the key findings resulting from this work.

SYSTEM OVERVIEW[5,6]

The Westinghouse CRD system performs an electromechanical conversion that enables low-level electronic signals generated by the rod control system to cause a change in the physical position of clusters of individual neutron-absorbing control rods, referred to as rod cluster control assemblies (RCCAs). A typical four-loop PWR employs 53 RCCAs individually positioned by dedicated CRDMs. CRDM operation and hence the position of a mechanically coupled RCCA depend on electromagnetic forces developed by an arrangement of coils that surround the assembly. Although magnetic forces are required to hold or reposition an RCCA, insertion is accomplished by the force of gravity acting on the weight of the assembly. Upon removal of electrical power from the CRDM at any time during its operating sequence, the force of gravity will cause the RCCA to drop into the core. The total insertion time is designed to be less than 2.2 s.

Rod position information is continuously monitored by two independent systems; the individual rod position indication system (IRPI) and the bank demand position indication system (BDPI) provide the operator with both an actual and an inferred indication of RCCA position.

The following paragraphs provide a brief description of each major subassembly bounded by the assessment of aging.

Rod Cluster Control Assembly

Each RCCA comprises a cluster of individual rods connected at one end to a common hub or "spider" assembly. The specific number of control rods that make up an RCCA will vary according to the size of the fuel assembly employed by the plant. Domestically operated Westinghouse PWRs may use either a 14 × 14, 15 × 15, or 17 × 17 fuel assembly array. In the 15 × 15 fuel assembly array, the fuel rods are arranged in a square pattern with 15 fuel locations on each side. The RCCA used with this assembly consists of 20 individual absorber rods positioned in mating guide thimbles that form an integral part of the fuel assembly and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. Similarly, the RCCA used with the 14 × 14 fuel array consists of 16 rods, whereas 24 rods are used in an RCCA for the 17 × 17 array. The alignment of absorber rods to guide thimbles is maintained by the overall length of the RCCA, which causes the tips of the rods to remain engaged in their mating guide thimbles when the assembly is fully withdrawn.

The most widely used neutron-absorbing material is an alloy of 80% silver, 15% indium, and 5% cadmium

(Ag–In–Cd). The Ag–In–Cd absorber is in the form of extruded rods sealed in type 304 stainless steel tubes. The individual absorber rods are suspended by cylindrical fingers of the RCCA spider assembly. Each rod is threaded into the spider fingers and pinned; the pins are then welded in place. In newer plants a mechanical lock may exist instead of a weld; this will allow easier replacement of individual rods. Radial vanes connect the spider assembly fingers to a central hub that permits mechanical coupling to the CRDM drive rod.

In addition to maintaining the geometric integrity of the core, structural internal reactor components play a vital role in ensuring that its reactivity can be appropriately controlled. Within the upper internals of the reactor, control-rod-guide-tube (CRGT) assemblies shield and guide the RCCA above the core. In the lower portion of the CRGT, sheaths and split tubes provide continuous lateral support for the control rods between approximately 22 and 40 in. above the upper core plate. Above this region guide plates provide intermittent lateral support for the control rods. Additional guidance for the drive rods of the CRDMs is provided by the upper extension of the guide tube that is attached to the upper support.[5] Note that variations in the design of the CRGT exist and depend on the specific design of the upper internals of the reactor.

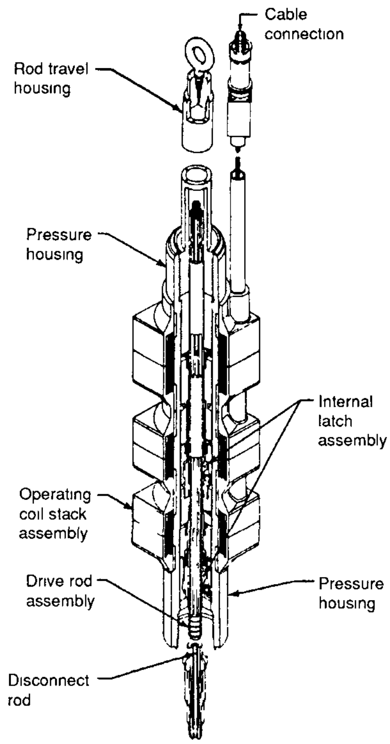

Fig. 2 Control-rod-drive mechanism.

Control-Rod-Drive Mechanisms[3–5]

A typical four-loop Westinghouse PWR employs 53 CRDMs located on the dome of the reactor vessel. Each CRDM is mechanically coupled to an RCCA. As depicted in Fig. 2, the major subcomponents of a CRDM are the pressure housing, drive rod assembly, internal latch assembly, and operating coil stack assembly.

The external pressure housings of the CRDM form part of the reactor coolant pressure boundary. Reactor coolant water fills the pressure housing and immerses all moving parts located within this enclosure. The upper part of the pressure housing, or rod travel housing, provides space for the drive rod during its upper movement as control rods are withdrawn from the core. The lower portion of the pressure housing, or latch housing, contains the latch assembly and is threaded and seal welded onto adapters located on top of the reactor pressure vessel. The latch housing and rod travel housing are connected by a threaded, seal-welded maintenance joint.

Westinghouse CRDMs operate as magnetically controlled jacks. Each CRDM is provided with an independent, air-cooled operating coil stack assembly comprising three electromagnetic coils (stationary gripper coil, movable gripper coil, and lift coil). The assembly is concentrically mounted on the CRDM and rests on the base of the latch housing without mechanical attachment. Each coil is energized in a controlled sequence by solid-state switches located in the power cabinet. Magnetic flux induced by the coils through the housing walls causes internal stationary or movable latching mechanisms to engage a grooved drive shaft that is mechanically coupled to an RCCA. Therefore sequential operation of the latch assembly enables the CRDM to hold, lift, or insert the RCCA. Two lead wires per coil are carried through a conduit to the top of the CRDM pressure housing where they terminate in a single six-pin electrical connector.

During normal, steady-state plant operation, the stationary gripper coil remains energized to hold its associated RCCA in a fixed position. The CRDM will hold the RCCA in this position until either (1) an insertion or withdrawal stepping sequence is initiated by the rod control system or (2) an interruption in electrical supply power causes the RCCA to fall, by gravity, into the core. Each sequential operation of the CRDM produces an incremental step motion of $^5/_8$ in. (insertion or withdrawal) of RCCA travel. The CRDM is capable of withdrawing or inserting an RCCA at a maximum speed of 72 steps (45 in.) per minute and develops a lifting force of approximately twice its static lifting load to provide extra capacity for overcoming mechanical friction.

Power Cabinets[6]

The power cabinets convert the 260-V, three-phase, a-c supply power fed from the reactor trip breakers to a direct current suitable for operation of the CRDM operating coil stack assembly. Its solid-state circuitry principally consists of phase-controlled, half-wave, thyristor bridge circuits that supply a pulse of direct current of a predetermined magnitude and duration to the operating coils of the CRDM. A typical four-loop plant employs five functionally identical power cabinets, each capable of supporting the operation of up to 12 CRDMs.

Logic Cabinet[5]

A single logic cabinet contains all the low-level electronic circuitry necessary to develop rod position command signals generated by the rod control system into the appropriate switching signals required for the proper sequencing of power conversion circuits located in the power cabinets. The design employs solid-state, integrated circuits mounted on plug-in printed circuit cards. Relays, required to drive auxiliary equipment, such as the plant computer and annunciators, are also located within this cabinet.

Rod Position Indication System[7]

Two independent systems monitor control-rod position: IRPI and BDPI. IRPI obtains information from detectors located on each CRDM and therefore presents the actual position of each RCCA. BDPI obtains its information by counting the number of steps of rod motion demanded by the rod control system. This information, although more accurate than that of the IRPI, is an inferred indication and represents the position that a group of rods should hold.

ASSESSMENT OF OPERATING EXPERIENCE

The assessment of component aging within the Westinghouse CRD system included a review of operational experience information obtained from three sources: the Nuclear Plant Reliability Data System (NPRDS), Licensee Event Reports (LERs), and Nuclear Power Experience (NPE). NPRDS and NPE data covered a nine-year period between Jan. 1, 1980, and Dec. 31, 1988, whereas the review of LER data encompassed reports submitted during a five-year period between Jan. 1, 1984, and Dec. 31, 1988. Each failure record was individually reviewed and encoded into a computerized data base developed specifically to avoid the possibility of double counting. The three data sources provided an average of 30 unique failure events per year over a ten-year period, of which approximately 35% were directly attributable to aging. When the failure reports were categorized in accordance with the subassembly in which the failure occurred, the majority of reported failures were in the electrical portion of the system, namely, the power and logic cabinets, interconnecting cabling and connectors, and the rod position indication subsystem (Fig. 3). The following paragraphs provide a brief description of the leading aging mechanisms, failure modes, and causes for each subassembly. Key findings are further summarized in Table 1 (Ref. 2).

RCCA and Neighboring Components

Significant stresses related to this assembly include vibration, radiation, cyclic fatigue, and friction between this assembly and its neighboring components. The synergistic action of these degradation mechanisms on control-rod cladding surfaces has resulted in three distinct types of RCCA wear: fretting wear, sliding wear, and intergranular stress corrosion cracking (IGSCC).

Hydraulically induced vibrations, developed by the flow of primary coolant up through the core, cause

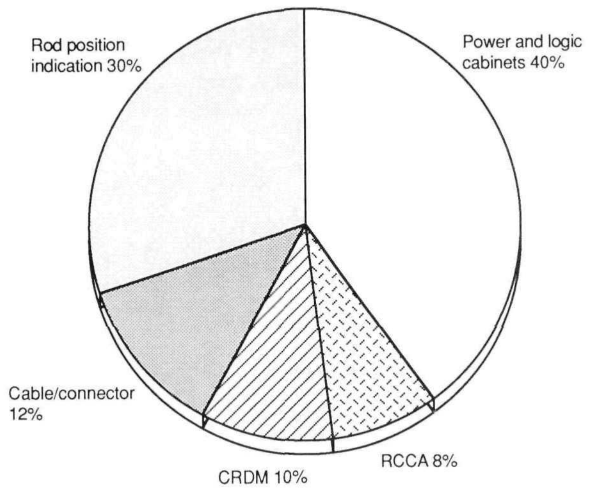

Fig. 3 Control-rod-drive system failures. RCCA, rod cluster control assemblies; CRDM, control-rod-drive mechanism.

Table 1 Summary of Dominant Component Failure Modes and Mechanisms

Subassembly	Component failure mode	Failure cause	Failure mechanism
Rod cluster control assembly (RCCA)	Abnormal or unexpected wear of absorber rod cladding	Mechanical wear and fatigue	Localized "fretting" caused by hydraulically induced contact with CR insertion "Sliding" wear caused by clad–guide block contact during RCCA stepping and trip insertion
		Absorber–clad contact caused by irradiation-induced swelling Ag–In–Cd absorber combined with mechanical fatigue and wear of type 304 stainless steel cladding	Synergistic stresses leading to intergranular stress corrosion cracking to type 304 stainless steel cladding material
	Degradation or cracking of Inconel X-750 CR guide-tube-support pin material	Inadequate heat treatment of Alloy X-750 (Inconel) material during fabrication	Stress corrosion cracking
	Spider assembly van weld failure	Mechanical–material fatigue	Normal wear–age
Control-rod-drive mechanism (CRDM)	Operating coil fails open	Normal wear–age	Conductor material fatigue and thermal stress
	Operating coil fails shorted	*Degradation of coil insulation*	*Environmental stress (steam–boric acid, heat)*
	Latch assembly misstepping	Buildup of foreign material	Small particle debris present in rod control system
	Latch assembly failure to withdraw	Mechanical binding	Thermal cycling
	Shroud cooling fan fails to operate	Mechanical wear, fatigue	Age–normal wear
Power and logic cabinets	Printed circuit boards provide incorrect output	Subcomponent failure	Environmental stress (heat and humidity) Vibration Voltage transients Age
		Connector pin	Corrosion–oxidation Vibration Mechanical wear
	Fuse fails open	Fusible link material fatigue	Age Vibration Cyclical load transients Heat
	Excessive voltage drop across fuse	Degraded condition of fuse contact material	Age Corrosion
	d-c power supply no output	Internal component failure	Environmental stress (heat) Voltage–current transient
		Trip of thermal protection circuits	Environmental stress (heat)
Cables–Connectors	CRDM coil stack connector fails open	Mechanical wear, fatigue	Normal wear–age
		Corrosion	Environmental stress (moisture–boric acid intrusion)
Rod position indication (RPI)	Incorrect RPI analog RPI	Instrument calibration drift	Thermal variation in CRDM during primary system heatup

repetitive contact between the control rod and adjacent guide blocks of the CRGT assembly located in the upper internals of the reactor. The mechanical stress caused by the physical impact of the rod with the guide cards forms localized areas of fretting wear on the cladding surface at points along the length of the control rod. Although this aging mechanism cannot be entirely eliminated, operating procedures that require a periodic change in the axial position of the RCCAs has been found to mitigate its effects by distributing the wear more evenly along the cladding surface.[8]

Rubbing the individual absorber rods against the guide blocks of the CRGTs during RCCA insertion and withdrawal has resulted in sliding wear of the absorber cladding. This type of wear is identified by long axial grooves worn into the cladding surface.

A combination of complex stress caused by mechanical interaction between the absorber material and cladding has been found to contribute to the development of IGSCC of the control-rod cladding. The mechanical interaction of the absorber and cladding apparently occurs as a result of absorber swelling caused by neutron exposure, exposure of the cladding to high levels of radiation, cyclic stresses induced by pressure variations in the primary system, and fatigue of the cladding caused by control-rod contact with fuel assembly guide thimbles.[9]

Stress corrosion cracking (SCC) failures of CRGT support (or split) pins used to maintain alignment and to support the guide tubes have been reported. SCC has been identified as a source of degradation in fuel-assembly, hold-down, spring clamp screws and CRGTs, including the split pins. This failure mechanism, caused by a high level of tensile stress, chemical environment, and material susceptibility to attack, typically causes the material to become embrittled and thus ductility is reduced. In addition to causing an RCCA to become stuck, split pin failures could lead to steam generator damage. Westinghouse has redesigned the split pins, including revising heat-treatment requirements and reducing the torque requirements of its mating cylindrical nut.

Control-Rod-Drive Mechanism

Because of its location and mode of operation, the CRDM is exposed to a number of potentially degrading stressors, including the high-temperature corrosive environment inside CRDMs, thermal transients that occur during plant heatups and cooldowns, mechanical forces developed during normal stepping movement of the control rods, radiation, and temperature. Parts of the CRDM exposed to the reactor coolant water, such as the latch assembly, drive rod, and pressure housing, are constructed from corrosion-resistant materials, including stainless steel, Alloy X-750 (Inconel), and cobalt-based alloys.

The CRDM housing forms part of the primary system pressure boundary. Therefore the principal stresses of concern for this assembly are those which can contribute to its rupture. Pressure housings are made from type 304 stainless steel, which may be cast or forged. Cast housings are a particular concern because of their susceptibility to thermal embrittlement, which may lead to the development of small cracks in the housing.[10] Such leaks would be difficult to detect and have the potential to expose neighboring components to additional chemical stress from exposure to the corrosive boric acid in the primary coolant.

The CRDM latch assembly contains fixed and movable magnetic pole pieces that actuate two sets of gripper latches. The principal operating stresses for this assembly include cyclic fatigue, the corrosive environment of the primary coolant, and mechanical binding and loading forces developed during the numerous insertion and withdrawal sequences that occur during the operating life of a CRDM. In addition, any small particles present in the coolant may become lodged in the latch assembly and thus cause mechanical binding, jamming, or misstepping of the mechanism.

The operating coil stack assembly is concentrically mounted over the CRDM and surrounds the latch housing. Its principal operating stresses include temperature, moisture (steam–boric acid), corrosion, and radiation. Thermal stress primarily affects the insulation quality of the operating coils and their connected cabling. The normally energized stationary gripper coils have experienced a much higher failure rate (order of magnitude) than the movable gripper and lift coils. Although it is not apparent why these coils are failing at a faster rate than other similarly constructed coils located on the same assembly, their nearly continuous operation is believed to induce ohmic heating in small, localized areas. The resulting temperature increase in these areas could contribute to their degradation.

Power and Logic Cabinets. Heat, humidity, supply voltage transients, vibration, and corrosion are the significant stresses related to this subassembly. An elevated level of ambient temperature was identified as the leading contributor to the failure of components within the power cabinet. Additionally, heat dissipated by certain components of this assembly, such as thyristors and transformers, increases the total thermal load.

As shown in Fig. 4, printed circuit logic cards accounted for nearly half of all problems related to the power and logic cabinet subassembly. The failure of these components has had a significant impact on plant performance, with 60% of the circuit board failures described in the LER data resulting in a reactor trip. Although the dominance of printed circuit card failures may be attributed to their relatively large population, the review of the operating experience data also identified high ambient temperatures in the vicinity of the cabinets to be a primary contributor to their failure, with certain circuit boards apparently more susceptible to failure than others located within the same cabinet. Specifically, "Firing Circuit" cards located within the power cabinet and "Slave Cycler Counter" cards of the logic cabinet were each found responsible for approximately 20% of the failures related to their respective assemblies. This failure rate was significantly higher than that of other cards located within the same assembly.

Fuses and their associated mounting hardware (i.e., "fuse clips") are also significant contributors to power and logic cabinet failures. Several instances of observed fuse degradation, such as "fatigue" or "normal end of life," were found in the operating experience data.

Cables and Connectors. Cables and connectors, which link the power cabinets to the CRDMs, accounted for approximately 12% of the system failures. Of those, problems with operating coil stack connectors, located at the CRDM, were the dominant contributor. Mechanical wear of the plug contacts and corrosion in the area of the connector pin-mating surfaces were the leading degradation mechanisms identified. All the reported failures occurred at plants in operation for more than five years. Coil stack connector problems have been recognized by Westinghouse, which now recommends replacing the connectors with a newer design.[11]

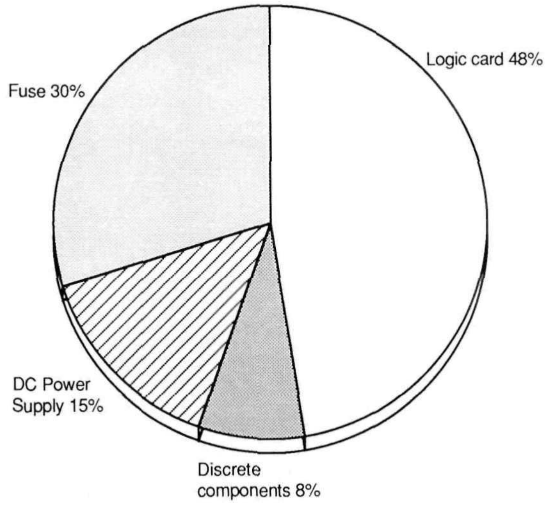

Fig. 4 Power and logic cabinet subcomponent failures.

Rod Position Indication. Depending on plant vintage, either an analog or digital type IRPI may be employed. The detector coils of both systems are mounted on the rod travel housing of the CRDM and therefore are subjected to many of the same environmental stresses described for the operating coil stack assembly. The consequences of detector coil failure are more significant in the analog type of IRPI. Because the detector used in this type of system is basically a linearly variable differential transformer (LVDT), a single open winding will initiate a false rod drop signal and cause a loss of all position information for the affected rod.

As shown in Fig. 3, the RPIS accounted for a large portion (30%) of the total number of reported failures. The majority of problems associated with this system were due to calibration drift errors of the analog type of IRPI. The calibration of this system appears to be influenced by temperature variations of the CRDM during heatup of the primary system.

The BDPI derives its information from step-counting circuits located in the logic cabinet. Therefore those stresses discussed previously for the power and logic cabinets may also impact the availability of this system. Additionally, this system is susceptible to mechanical wear and cyclic fatigue of its solenoid-driven electromechanical counters.

In general, RPI failures were not observed to have a significant impact on plant operation. However, the potential for analog IRPI systems to display misleading rod position information cannot be overlooked. Additionally, this problem appears to be common to multiple plants, which may indicate a need for a generic resolution.

DETECTING AND MITIGATING DEGRADATION

Appropriate preventive maintenance (PM) and monitoring techniques are key tools for detecting and mitigating age-related degradation. With the cooperation and support of industry, the Electric Power Research Institute, and the Nuclear Management and Resources Council, a survey of inspection, surveillance, monitoring, and maintenance activities that could be used to detect and mitigate degradation of CRD system components was accomplished.[2] The purpose of this survey was to obtain information and insights from as many specific plants as possible, including more detailed information on system modifications, maintenance practices, and operation. Representing 10 utilities, 15 plants provided information on PM practices, design changes to reduce stresses imposed on the system, and some advanced techniques for monitoring system performance.

Survey responses related to PM programs for mechanical and electrical components are summarized in Tables 2 and 3, respectively. As is readily apparent from a comparison of these tables, plants responding to the survey indicate that they expend the largest portion of their available PM resources on the electrical portion of the system. Although the survey results indicate that nearly all plants perform some form of routine inspection of the electrical components located within containment, such as the CRDM coil stack assembly, cables, and connectors, only one plant performs a visual inspection of the CRDM drive rod on a scheduled basis.

The increased resources allocated to the electrical portion of the CRD system are certainly supported by operational experience data, which showed electrical components to be a leading contributor of problems.

Table 2 Preventive Maintenance for Mechanical Components

Component	Maintenance	No. of plants involved and frequency[a]
Rod cluster control assembly–guide tube	Eddy-current testing	3
	Profilometry	1
Latches	None	Not applicable
Drive rod	Visual inspection for wear and crud buildup	1
Seal welds	Nuclear power experiences	3 @ 10% every 10 yr
	Hydrostatic	6
	Remote	2
Vent valves–plugs	Hydrostatic	4
Ventilation	Clean and inspect	All
	Lube fan bearings	6
	Megger motors	8

[a]Intervals are once per refueling cycle unless otherwise noted

Table 3 Preventive Maintenance for Electrical Components

Component	Maintenance	No. of plants involved and frequency[a]
Coil stack assembly	Insulation resistance	7
	Coil resistance	14
	Coil timing signature traces	4
	Polarity check	3
Cable in containment	Insulation resistance	7
	Visual	6
Connectors in containment	Inspect for tightness	14
	Check watertight seal	2
Rod position indication wiring in containment	Visual	4
	Resistance measurements	10
	Insulation resistance	3
Cables–connectors outside containment	Insulation resistance	7
	Resistance measurements	3
Power supplies	Visual inspection	3
	Calibrate protective devices	5
	Vendor refurbishment	1
	Functional test	8
Power, control, and logic cabinets	Visual inspection	5
	Vendor refurbishment	4
	Replace fuses	4 @ 3 yr
	Measure cabinet temperature	4
	Timing–functional test	2

[a]Intervals are once per refueling cycle unless otherwise noted

However, in spite of increased attention, the failure rate of certain electrical components, such as operating coil stack cable connectors and printed circuit logic cards, remains relatively high when compared with other components of the system. Although the relatively large number of reported electrical component problems may be attributed to their population, it may also be indicative of design deficiencies of certain components or a need for improved maintenance and monitoring techniques.

Plant modifications (i.e., design changes) are a proven method of mitigating the effects of age: moreover, for certain cases, such as the analog IRPI system problems described earlier, they may be the only long-term solution. However, cost-effective resolutions may frequently be achieved through improved PM techniques. For example, corrosion caused by moisture and/or boric acid intrusion has been identified as a common cause of coil stack connector degradation. The failure of this component typically has a significant effect on system performance and plant operability, which often results in a stuck control rod or rod drop event Westinghouse has recognized the relatively high incidence of coil stack connector problems and has recommended the replacement of these devices with a newer design [11] Of the 15 plants responding to the survey, 3 indicated that they had replaced or planned to replace these devices The survey results also indicate, however, that, although the vast majority of plants routinely inspect CRDM coil stack connectors for signs of degradation, only 2 of the 15 regularly check the watertight seal

Improved PM of the power and logic cabinets also appears to warrant further consideration As stated earlier, this subassembly accounted for approximately 40% of all CRD system problems Of those, printed circuit logic card failures were found to dominate the data These failures have also had a significant effect on plant operation, with 60% resulting in a reactor trip Although the primary failure mechanism for these devices appears to be thermal stress caused by high ambient temperature, only four plants responding to the survey regularly monitor cabinet temperature

A relatively large variation among plants was also noted in PM activities associated with the mechanical portion of the system Although more than half the plants responded positively to indications of RCCA and guide-tube wear, only four were found to perform periodic testing of these components

The survey also asked for the utility's opinion regarding the monitored parameters, tests, or inspections that are most important for ensuring the operational readiness of the CRD system The results clearly revealed the opinion that testing required by the Technical Specifications (rod drop timing and rod exercise test) is useful for determining system performance

CONCLUSIONS AND RECOMMENDATIONS

The results of an industry survey found that, in the opinion of the majority of plants responding, testing required by the plant Technical Specifications is useful for determining system performance As evidenced in the review of historical operational data, however, undetected age-related degradation has occurred and, in several cases, has led to component failures that have initiated unnecessary challenges to the safety protection features of the plant Such occurrences indicate that the timely replacement of degraded components before failure is not always possible with existing condition monitoring techniques Therefore improvements in preventive maintenance, condition monitoring, and design warrant additional consideration to more effectively detect and mitigate the effects of aging in the Westinghouse CRD system

The goals of plant design modifications should be directed at (1) reducing or eliminating the stresses that contribute to aging degradation or (2) improving the materials to better withstand existing stresses Specific examples of recommended modifications that should be considered include replacement of the CRDM operating coil stack connectors, upgrade from an analog RPIS to a digital multiplexing design, improved ventilation for rod drive cabinets and CRDMs, upgrade of cables located in containment to higher temperature-rated assemblies, and installation of permanently installed test equipment to minimize the potential for human error and to mitigate the influence of testing on aging (e g , disconnecting–reconnecting) Industry survey results indicate that some facilities have successfully mitigated the effects of aging on certain components of the CRD system through implementing such plant modifications

Advanced methods of monitoring CRD system integrity will permit early detection of its degradation Existing monitoring techniques worthy of further consideration for applicability to this system include the following

- *Underwater TV cameras can be used to conduct and document a thorough, visual inspection of normally inaccessible mechanical and structural components* This assessment could be conducted periodically to document the condition of critical parameters, such as wear of the RCCA cladding and guide tube and CRDM drive rod and latches Areas of observed wear could be noted and their cause determined In addition, periodic visual inspections could be supplemented with one or more of the more advanced monitoring techniques currently employed at some plants, such as ultrasonics, eddy current, and profilometry

- *The American Society of Mechanical Engineers inservice inspection requirement specifies that welds on 10% of the peripheral CRD housings be inspected* This does not completely assess the weld integrity of the interior housings Nil ductility transition techniques and equipment should be developed to allow for the remote inspection of the interior housings as well

- *A current signature analysis of CRD system performance, currently being used by one plant, should be evaluated by other utilities* This technique, which monitors CRDM operating coil current during rod motion, provides data necessary to determine the acceptability of power and logic circuitry and coil integrity In addition, a rough indication of mechanical interferences can be ascertained

• *The review of operating experience data identified CRDM coil stack assembly cables and connectors located in containment as leading contributors of system failures* For the detection of the degradation of these circuits before failure, the common maintenance practice of measuring the resistances of cables, connectors, and coils at regulated conditions (baseline temperature) following each refueling outage is recommended In addition, more accurate monitoring methods that are commercially available, such as the Electronic Characterization and Diagnostics System (ECAD), should be considered By performing a precise examination of important operational parameters, such as coil resistance, insulation resistance, capacitance, and dissipation factor, these systems are capable of providing an early indication of circuit degradation For example, one plant reported that, through the use of ECAD, a 0.8-ohm difference in the resistance of one coil was detected This reduction in resistance, which was also detected as a lower inductance, indicated the potential for a short circuit in the winding of the affected coil. Additionally, by monitoring the capacitance and dissipation factor, such systems are capable of detecting moisture intrusion, which is a common cause of cable connector problems

• *Component failures within the power and logic cabinets contributed to the majority (40%) of reported problems with the Westinghouse CRD system* Of those, printed circuit logic card failures were found to dominate the data. The failure of printed circuit logic cards also had a significant effect on plant operation, with 60% resulting in a reactor trip Although the primary failure mechanism for these devices appears to be thermal stress caused by high ambient temperature, only four plants responding to the industry survey regularly monitor cabinet temperature Clearly, increased attention to ambient temperature of the power and logic cabinets is warranted. Further, more frequent testing of the printed circuit cards to manufacturer specifications (particularly the power cabinet "Firing Circuit" and logic cabinet "Slave Cycler Counter" cards) should be considered

Other functional indicators of possible value in detecting degradation in the CRD system include the following.

• *CRDM cycle counter* Because the amount of wear on components (such as the drive rod and latch assembly) is directly related to rod movement, it is recommended that the number of control-rod steps for each control bank of RCCAs be counted

• *Operating experience* With essentially 50 plants using the same basic CRD system design, tremendous benefits can be derived from shared information. Time-dependent degradation experienced at older plants, for example, should influence the maintenance practices at other plants

REFERENCES

1 J P Vora, *Nuclear Plant Aging Research (NPAR) Program Plan Components Systems and Structures* Report NUREG-1144, Rev 1, September 1987

2 W Gunther and K Sullivan, *Aging Assessment of the Westinghouse PWR Control Rod Drive System*, Report NUREG/CR-5555 (BNL-NUREG-52232), March 1991

3 *Final Safety Analysis Report* Consolidated Edison Company of New York, Indian Point 2, Docket No 50 247

4 *Final Safety Analysis Report*, Sequoyah Units 1 and 2, Docket Nos 50-327 and 50-328

5 A G Ware Pressurized Water Reactor Control Rod Mechanisms and Reactor Internals, Chapter 7 of *Residual Life Assessment of Major Light Water Reactor Components Overview*, Report NUREG/CR-4731-Vol 2 (EGG-2469-Vol 2), November 1989

6 A Blanchard and D N Katz, *Solid State Rod Control System—Full Length*, Report WCAP-7778, January 1972

7 A E Blanchard, *Topical Report—Rod Position Monitoring*, Report WCAP-7571

8 P Sipush et al , *Lifetime of PWR Silver–Indium–Cadmium Control Rods*, Report EPRI NP 4512, Electric Power Research Institute, March 1986

9 E J Brown, *Localized Rod Cluster Control Assembly Wear at PWR Plants*, Report AEOD Engineering Evaluation Report E613, December 1986

10 *PWR Pilot Life Extension Study at Surry Unit 1 Phase 2*, Report EPRI NP-6232-M, Electric Power Research Institute, March 1989

11 Correspondence from W J Johnson to S K Aggarwal, Westinghouse Comments Regarding Draft NUREG/CR-5555, NS-NRC 90-3531, Nov 21, 1990

Reactor Shutdown Experience

Compiled by J. W. Cletcher[a]

This section presents a regular report of summary statistics relating to recent reactor shutdown experience The information includes both numbers of events and rates of occurrence. It was compiled from data about operating events entered into the SCSS data system by the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory and covers the three-month period of April, May, and June 1991. Cumulative information, starting from May 1, 1984, is also shown. Updates on shutdown events included in earlier reports are excluded.

Table 1 lists information on shutdowns as a function of reactor power at the time of the shutdown for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). Only reactors in commercial operation at the start of the reporting period (Apr. 1, 1991) are included. The second column for each reactor type shows the annualized shutdown rate for the reporting period. The third and fourth columns list cumulative data (numbers and rates) starting as of May 1, 1984

Table 2 shows data on shutdowns by shutdown type· *Real Scrams* are events in which the reactor was scrammed for a valid cause, *Spurious Scrams* are events in which an instrument failure or other fault causes a scram not actually called for by existing reactor conditions, *Non-Scram Shutdowns* (frequently from operating power to hot standby) do not involve actuation of the scram system either manually or automatically. Only reactors in commercial operation are included The second column for each type of reactor

Table 1 Reactor Shutdowns by Reactor Type and Percent Power at Shutdown[a] (Period Covered is the Second Quarter of 1991)

	BWRs (37)				PWRs (75)			
Reactor power (P), %	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
0	11	1 19	589	2 39	7	0 37	384	0 79
$0 < P \leq 10$	0	0 00	109	0 44	2	0 11	146	0 30
$10 < P \leq 40$	6	0 65	134	0 54	8	0 43	278	0 57
$40 < P \leq 70$	3	0 33	117	0 47	3	0 16	148	0 31
$70 < P \leq 99$	4	0 43	291	1 18	8	0 43	422	0 87
$99 < P \leq 100$	13	1 41	323	1 31	15	0 80	872	1 80
Total	37	4 01	1563	6 33	43	2 30	2250	4 64

[a]Data include shutdowns for all reactors of the designated type while in commercial service during all or part of the period covered The cumulative data are based on the experience while in commercial service since the starting date of Jan 1, 1984, through the end of the reporting period, it includes the commercial service of reactors now permanently or indefinitely shut down

[b]Based on cumulative BWR operating experience of 246 76 reactor years

[c]Based on cumulative PWR operating experience of 484 57 reactor years

[a]Oak Ridge National Laboratory

shows the annualized rate of shutdowns for the reporting period. Cumulative information is shown in the third and fourth columns for each reactor type

Table 3 lists information about shutdowns by reactor age category, both total numbers and rates in that category, it also shows cumulative results. Note that the age groups are not cohorts, rather reactors move into and out of the specified age groups as they age. The reactor age as used in this table is the number of full years between the start of commercial operation and the beginning of the reporting period (Apr. 1, 1991, for this issue) The first line of this table gives the information for reactors licensed for full power but not yet in commercial operation on that date.

Table 2 Reactor Shutdowns by Reactor Type and Shutdown Type[a] (Period Covered is the Second Quarter of 1991)

	BWRs (37)				PWRs (75)			
Shutdown (SD) type	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
SDs required by Technical Specifications	10	1 08	199	0 81	4	0 21	337	0 70
Intentional or required manual reactor protection system actuations	7	0 76	126	0 51	8	0 43	260	0 54
Required automatic reactor protection system actuations	12	1 30	726	2 94	23	1 23	1254	2 59
Unintentional or unrequired manual reactor protection system actuations	0	0 00	9	0 04	0	0 00	18	0 04
Unintentional or unrequired automatic reactor protection system actuations	8	0 87	503	2 04	8	0 43	381	0 79
Total	37	4 01	1563	6 33	43	2 30	2250	4 64

[a]Data include shutdowns for all reactors of the designated type while in commercial service during all or part of the period covered The cumulative data are based on the experience while in commercial service since the starting date of Jan 1, 1984, through the end of the reporting period, it includes the commercial service of reactors now permanently or indefinitely shut down

[b]Based on cumulative BWR operating experience of 246 76 reactor years

[c]Based on cumulative PWR operating experience of 484 57 reactor years

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age[a]
(Period Covered is the Second Quarter of 1991)

	BWRs (37)						PWRs (75)					
Years in commercial operation (C.O.)	Exposure during the period (in reactor years)	Number: Reactors	Number: Shutdowns	Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year	Exposure during the period (in reactor years)	Number: Reactors	Number: Shutdowns	Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year
Not in C O[b]	0 249	1	0	0 00	330	28 75	0 000	0	0	0 00	334	35 21
First year of C O	0 000	0	0	0 00	121	9 00	0 498	2	1	2 01	273	10 13
Second through fourth year of C O	1 246	5	3	2 41	250	6 68	2 245	10	5	2 23	467	5 84
Fifth through seventh year of C O	1 744	7	10	5 73	105	4 85	2 987	13	8	2 68	225	3 90
Eighth through tenth year of C O	0 498	2	1	2 01	147	7 07	2 491	10	4	1 61	314	4 71
Eleventh through thirteenth year of C O	0 249	1	1	4 01	267	6 00	1 177	5	2	1 70	440	4 58
Fourteenth through sixteenth year of C O	1 246	5	2	1 61	370	6 40	2 705	13	9	3 33	307	3 48
Seventeenth year and over	4 235	17	20	4 72	303	5 92	6 582	27	14	2 13	210	3 16
Total	9 467		37	3 91	1893	7 33	18 686		43	2 30	2580	5 22

[a]Age is defined to be the time (in years) from the start of commercial operation to the time of the shutdown event, except for the first line, which lists reactors not yet in commercial service (see b below)

[b]This category includes reactors licensed for full-power operation but not yet commercial During this reporting period reactors in this category included 1 BWR (Shoreham) and no PWRs

Selected Safety-Related Events

Compiled by G. A. Murphy[a]

LOSS OF SPENT FUEL POOL WATER AT WOLF CREEK GENERATING STATION[1]

On Sept. 23, 1991, Wolf Creek Generating Station (WCGS)[b] was in cold shutdown (Mode 5) with the reactor coolant system (RCS) in a solid condition (hydrostatically full). Four reactor coolant pumps (RCPs) were running. The fuel transfer canal was partially filled and was being used as a holding reservoir for borated water. The fuel transfer canal gate was in place, and the dual boot seals were inflated. The fuel transfer tube was closed at both ends of the containment penetration. The licensee was preparing to perform maintenance on the non-safety-related startup (SU) transformer—one of two power sources to non-safety-related bus PA01. The SU transformer was already de-energized, and bus PA01 was being supplied from an alternate power source via circuit breaker PA01-01 and the unit auxiliary transformer. In accordance with plant procedures, personnel were verifying the hanging of clearance tags before starting work on the SU transformer. At 10:32 a.m., when the door of the cubicle for circuit breaker PA01-10 was closed, a stuck breaker relay actuated. This actuation caused circuit breaker PA01-01 to trip open and thus de-energize bus PA01.

The loss of bus PA01 caused the loss of two RCPs, two service water (SW) pumps, and other miscellaneous loads. In addition, and unknown to the operators, pneumatic isolation valve KA-PV011, which feeds the service air (SA) system from the instrument air (IA) system, failed closed, as designed.

The operators took immediate action to stabilize the RCS pressure and temperature with only two RCPs running. They then started two essential SW pumps to compensate for the loss of the SW pumps and attempted to re-energize bus PA01. When breaker PA01-01 would not reclose, the operators notified both the plant electricians and instrumentation and control (I&C) technicians. The operators also instructed personnel to stop all work in the switchyard to ensure that no other power interruptions would occur.

At 11:54 a.m., the operators received a report of an odor in the vicinity of the B spent fuel pool (SFP) cooling pump (the odor was later determined to be from building floor drains because of the ventilation system imbalance caused by the bus loss). An operator was sent to the SFP cooling pump room. During this time the operators were still monitoring their control boards and, because of the reported odor, the SFP cooling pump in particular. While checking the running status of the SFP cooling pump, the operators noted that the amber light for the B SFP cooling pump was lit instead of the expected red running light (the amber light signified that the pump had tripped). They also noted that the SFP level indicator, which was located near the pump control switch, was indicating off-scale low. An operator was sent immediately to the fuel building to visually observe the SFP water level. The operator reported that the SFP level appeared to be about 6 ft low and, after some investigation, determined that the SFP water was draining to the fuel transfer canal through the gate between the SFP and the fuel transfer canal.

Upon receiving this report from the operator in the fuel building, the control room operators determined that IA to SA isolation valve KA-PV011 had closed as a result of the bus PA01 power failure. Since the SA system depended on the IA system for its air supply (it had no air receivers and only one compressor, which was powered off bus PA01), this isolation combined with air leaks in the system caused the gate boot seals to deflate. The operators then initiated a fuel building isolation signal, restricted personnel access to the fuel building, and bypassed valve KA-PV011. The operators had no indication of an increase in radiation or in airborne radioactivity in the fuel building. The bypassing of valve KA-PV011 provided air to reinflate the gate boot seals and stop the loss of water from the SFP.

The operators commenced refilling the SFP from the refueling water storage tank (RWST). Because of

[a]Oak Ridge National Laboratory

[b]Wolf Creek is an 1170-MW(e) Westinghouse pressurized-water reactor operated by Wolf Creek Nuclear Operating Corporation and located near Burlington, Kans.

concerns about the RWST inventory and the erroneous assumption that the SFP level was in accordance with the Technical Specifications (TS) limiting condition for operation of 23 ft above the fuel, the operators stopped filling the SFP. About four minutes later, an alternate power lineup was established that restored power to the SFP level instrumentation. Upon noting the indicated level in the SFP of about minus 44 in. (or about 21.8 ft above the fuel), the operators commenced refilling the SFP from the chemical and volume control system blending tee (the normal filling source). However, when they determined that the resultant makeup flow rate was low and that this process would take an inordinate amount of time, they reestablished SFP filling via the RWST.

About this time the electricians and I&C technicians determined that the reason circuit breaker PA01-01 had failed to close was that the contacts for a synchronizing check relay, a permissive for this breaker closure, were open. Upon manually actuating this relay to close the contacts, circuit breaker PA01-01 was closed and power was restored to bus PA01.

The operators continued to refill the SFP and at 3:23 p.m. had raised the level above the TS-required limit of 23 ft above the fuel. The SFP level had been below the TS limit about three hours and forty minutes. When the SFP level was raised sufficiently, the A SFP cooling pump was restarted to restore pool cooling. The A pump, rather than the formerly running B pump, was selected because there was some concern that the tripping of the B pump on low SFP level may have resulted in a partially drained suction pipe. During the three hours and forty minutes that the SFP cooling was lost, the SFP temperature remained steady at about 83 to 86°F.

Nuclear Regulatory Commission Region IV sent an Augmented Inspection Team (AIT) to WCGS on September 25 to gather information regarding licensee actions and to review plant response to the event.

One identified cause of the SFP loss of water was a design problem with the hoses and fittings that supplied air to the gate boot seal. The design drawing was inconsistent in that it showed quick-connect fittings at both the SA side and the boot seal side; however, a note on the same drawing stated that only the boot seal side had the quick-connect fittings and that the SA side was to have a hard-pipe fitting. The licensee had issued, as a corrective action, a temporary modification calling for hard-pipe fittings at both ends of the rubber hoses. This should improve the capability of the air system to supply the boot seals. The AIT was informed, however, that the air leaks were not isolated to only the quick-connect fittings. The check valves in the SA line to the seal, which are intended to prevent loss of seal air pressure if an SA system failure occurs, were in such condition that they would have allowed air to leak back into the SA system. In addition, the SA isolation valves apparently had excessive packing leakage. The lack of conformance to design and the poor mechanical condition of the SA system check valves significantly contributed to the loss of seal integrity.

The failure of the seal (i.e., a complete seal failure) with the transfer canal empty, the fuel transfer tube open, the reactor cavity drains open, and the reactor vessel to reactor cavity seal not in place would have resulted in draining the SFP down to about 1 ft above the stored fuel. This scenario apparently had not been considered in the design-basis accident (DBA) analysis. For the WCGS DBA, only a loss of SFP water to either the fuel transfer canal (with the transfer tube closed) or the cask loading pool, which results in a minimum water level of no less than 10 ft above the stored fuel, had been considered. The licensee also conducted certain activities (e.g., preventive maintenance on the fuel transfer trolley) when the fuel transfer canal and the reactor cavity were drained and the fuel transfer tube was open. In addition, the potential existed that a loss of the SFP level during the reconstitution or inspection of fuel assemblies could result in the exposure of fuel.

REFERENCE

1. Condensed from NRC Inspection Report No. 50-482/91-28.

Operating U.S. Power Reactors

Compiled by M. D. Muhlheim[a] and E. G. Silver[a]

This update, which appears regularly in each issue of *Nuclear Safety,* surveys the operations of those power reactors in the United States which have been issued operating licenses. Table 1 shows the number of such reactors and their net capacities as of June 30, 1991, the end of the three-month period covered in this report. Table 2 lists the unit capacity and forced outage rate for each licensed reactor for each of the three months covered in each report and the cumulative values of these parameters at the end of the covered quarter since the beginning of commercial operation. The information for this table was obtained from the Nuclear Regulatory Commission (NRC) Office of Information Resources Management. The Maximum Dependable Capacity (MDC) Unit Capacity (in percent) is defined as follows: (Net electrical energy generated during the reporting period × 100) divided by the product of the number of hours in the reporting period and the MDC of the reactor in question. The forced outage rate (in percent) is defined as: (The total number of hours in the reporting period during which the unit was inoperable as the result of a forced outage × 100) divided by the sum (forced outage hours + operating hours).

Table 3 and Fig. 1 summarize the operating performance of the U.S. power reactors during the three months covered by this report (April, May, and June 1991) and for the years 1989 and 1990.

In addition to the tabular data, this article discusses other significant occurrences and developments that affected licensed U.S. power reactors during this reporting period. It includes, but is not limited to, changes in operating status, regulatory actions and decisions, and legal actions involving the status of power reactors. We do not have room here for routine problems of operation and maintenance, but such information is available at the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20555.

Some significant operating events are summarized elsewhere in this section, and, when appropriate, a report on activities relating to facilities still in the construction process is given in an article "Status of Power-Reactor Licensing Activities" in the last section

Table 1 Licensed U.S. Power Reactors as of June 30, 1991

Status	No.	Capacity,[a] MW(e) (net)
In commercial operation[b]	112	100 269
In power ascension phase[c]	0	0
Licensed to operate at full power	112	100 269
Licensed for fuel loading and low-power testing[d]	0	0

[a]Based on maximum dependable capacity (MDC) where available, design electrical rating (DER) is used when the MDC rating is not available.

[b]Excludes Dresden 1 (DER = 200), Humboldt Bay (DER = 65), Three Mile Island 2 (DER = 906), LaCrosse (DER = 50), Fort St. Vrain (DER = 330), and Shoreham (DER = 820), all of which have operating licenses but are shut down indefinitely or permanently

[c]None at this time.

[d]None at this time

[a]Oak Ridge National Laboratory.

Table 2 Summary of Operating U.S. Power Reactors as of June 30, 1991[a]

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power MW(t)	Design power MW(e)	Commercial operation date	MDC unit capacity, % April	MDC unit capacity, % May	MDC unit capacity, % June	MDC unit capacity, % Cumulative (lifetime) thru 6/30/91	Forced outage rate, % April	Forced outage rate, % May	Forced outage rate, % June	Forced outage rate, % Cumulative (lifetime) thru 6/30/91
ARKANSAS 1 and 2, Pope County, Ark	50-313	PWR (B&W)	2568	850	12/74	101 0	97 2	99 2	58 4	0 0	2 4	0 0	13 0
(Arkansas Power & Light Co)	50-368	PWR (CE)	2815	912	3/80	0 0	103 8	98 6	69 6	0 0	0 0	0 0	12 3
BEAVER VALLEY 1 and 2, Shippingport, Pa	50 334	PWR (West)	2652	852	10/76	97 4	0 0	0 0	56 2	0 0	0 0	0 0	16 3
(Duquesne Light Co)	50-412	PWR (West)	2660	836	11/87	96 6	96 5	89 2	74 0	0 0	0 0	0 0	4 4
BIG ROCK POINT, Charlevoix County, Mich (Consumers Power Co)	50 155	BWR (GE)	240	72	3/63	99 2	97 6	36 7	60 9	0 0	7 0	62 8	7 9
BRAIDWOOD 1 and 2, Braidwood, Ill	50-456	PWR (West)	3425	1120	7/88	0 0	18 3	91 3	59 9	0 0	0 0	0 0	14 1
(Commonwealth Edison Co)	50-457	PWR (West)	3425	1120	10/88	99 1	61 5	88 5	72 4	0 0	27 8	0 7	4 0
BROWNS FERRY 1, 2, and 3, Decatur, Ala	50-259	BWR (GE)	3293	1065	8/74	0 0	0 0	0 0	33 9	100 0	100 0	100 0	55 0
(Tennessee Valley Authority)	50-260	BWR (GE)	3293	1065	3/75	0 0	0 0	0 0	32 1	100 0	100 0	100 0	55 3
	50-296	BWR (GE)	3293	1065	3/77	0 0	0 0	0 0	31 4	100 0	100 0	100 0	59 4
BRUNSWICK 1 and 2, Brunswick County, N C	50 325	BWR (GE)	2436	821	3/77	71 9	77 0	96 0	53 5	21 3	19 5	0 0	15 7
(Carolina Power & Light Co)	50-324	BWR (GE)	2436	821	11/75	88 2	69 8	92 7	50 3	8 8	25 4	0 9	13 6
BYRON 1 and 2, Byron, Ill	50-454	PWR (West)	3425	1120	9/85	91 7	98 5	80 9	72 0	0 0	0 0	0 0	3 0
(Commonwealth Edison Co)	50-455	PWR (West)	3425	1120	8/87	95 6	93 0	88 2	66 7	0 0	0 0	0 0	3 0
CALLAWAY 1, Callaway County, Mo (Union Electric Co)	50-483	PWR (West)	3411	1171	12/84	102 9	102 3	100 2	81 0	0 0	0 0	0 0	3 2
CALVERT CLIFFS 1 and 2, Lusby, Md	50-317	PWR (CE)	2560	845	5/75	103 8	56 6	0 0	66 2	0 0	0 0	0 0	9 5
(Baltimore Gas & Electric Co)	50-318	PWR (CE)	2560	845	4/77	0 0	56 1	55 9	67 7	0 0	24 6	41 1	5 7
CATAWBA 1 and 2, Lake Wylie, S C	50-413	PWR (West)	3411	1145	6/85	53 1	0 0	28 1	65 6	12 8	0 0	13 3	12 4
(Duke Power Co)	50-414	PWR (West)	3411	1153	8/85	100 7	96 0	100 2	66 4	0 0	3 3	0 0	13 4
CLINTON 1, Clinton, Ill (Illinois Power Co)	50-461	BWR (GE)	2894	933	11/87	56 6	97 2	96 0	52 4	27 6	0 0	0 0	15 8
COMANCHE PEAK 1, Glen Rose, Tex (Texas Utilities Electric Co)	50-445	PWR (West)	3411	1150	8/90	50 8	7 2	92 9	58 1	14 4	84 4	0 0	14 9
COOK 1 and 2, Benton Harbor, Mich (Indiana &	50-315	PWR (West)	3250	1030	8/75	99 8	66 6	89 1	65 7	0 0	15 6	4 5	7 0
Michigan Electric Co)	50-316	PWR (West)	3391	1100	7/78	90 4	98 5	95 0	61 5	7 7	0 0	0 0	13 0

(Table continues on the next page.)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		April	May	June	Cumulative (lifetime) thru 6/30/91	April	May	June	Cumulative (lifetime) thru 6/30/91
COOPER, Nemaha County, Nebr (Nebraska Public Power District)	50-298	BWR (GE)	2831	778	7/74	79 7	91 6	97 2	62 6	0 0	0 0	0 0	4 6
CRYSTAL RIVER 3, Crystal River, Fla (Florida Power Corp)	50-302	PWR (B&W)	2560	825	3/77	97 6	93 4	98 4	57 7	0 0	0 0	0 0	19 6
DAVIS-BESSE 1, Ottawa County, Ohio (Toledo Edison Co)	50-346	PWR (B&W)	2772	906	7/78	100 1	100 1	98 8	47 2	0 0	0 0	0 0	25 8
DIABLO CANYON 1 and 2, Diablo Canyon, Calif (Pacific Gas & Electric Co)	50-275 50-323	PWR (West) PWR (West)	3338 3411	1086 1119	5/85 3/86	0 0 96 3	87 4 95 8	100 1 98 9	73 9 77 7	0 0 0 0	10 2 0 0	0 0 0 0	4 1 5 3
DRESDEN 2 and 3, Grundy County, Ill (Commonwealth Edison Co)	50-237 50-249	BWR (GE) BWR (GE)	2527 2527	794 794	6/70 11/71	26 4 88 3	90 5 65 4	58 6 54 6	58 5 57 7	57 0 0 0	0 0 0 0	7 8 0 0	10 9 11 4
DUANE ARNOLD, Cedar Rapids, Iowa (Iowa Electric Light & Power Co)	50-331	BWR (GE)	1593	538	2/75	93 9	91 8	63 6	57 3	0 0	0 0	9 4	13 5
FARLEY 1 and 2, Dothan, Ala (Alabama Power Co)	50-348 50-364	PWR (West) PWR (West)	2652 2652	829 829	12/77 7/81	23 9 100 7	19 9 99 1	92 1 99 4	72 7 81 9	0 0 0 0	9 6 0 0	3 8 0 0	7 3 4 0
FERMI-2, Newport, Mich (Detroit Edison Co)	50-341	BWR (GE)	3292	1093	1/88	74 3	0 0	24 7	56 5	5 0	0 0	11 1	10 8
FITZPATRICK, Oswego, N Y (Power Authority of State of N Y)	50-333	BWR (GE)	2436	821	7/75	26 0	21 9	0 0	66 0	61 8	78 3	100 0	11 5
FORT CALHOUN, Washington County, Nebr (Omaha Public Power District)	50 285	PWR (CE)	1420	478	6/74	68 1	72 2	81 1	65 8	0 0	0 0	0 0	3 5
GINNA, Ontario, N Y (Rochester Gas & Electric Corp)	50-244	PWR (West)	1520	490	7/70	61 7	54 4	101 2	74 0	0 0	1 3	0 0	6 2
GRAND GULF 1, Port Gibson, Miss (Mississippi Power & Light Co)	50-416	BWR (GE)	3833	1250	7/85	103 7	74 6	72 5	73 6	0 0	18 9	21 2	6 3
HADDAM NECK, Haddam Neck, Conn (Connecticut Yankee Atomic Power Co)	50 213	PWR (West)	1825	582	8/67	54 4	101 0	89 5	75 3	40 0	0 0	0 0	5 9
HATCH 1 and 2, Baxley, Ga (Georgia Power Co)	50-321 50-366	BWR (GE) BWR (GE)	2436 2436	777 795	12/75 9/79	97 8 57 7	98 6 0 0	96 2 82 0	63 3 63 7	0 7 0 0	0 0 0 0	0 0 0 0	13 2 7 8

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		April	May	June	Cumulative (lifetime) thru 6/30/91	April	May	June	Cumulative (lifetime) thru 6/30/91
HOPE CREEK, Salem, N J (Publıc Servıce Electrıc & Gas Co)	50-354	BWR (GE)	3293	1067	12/86	91 9	86 3	99 2	78 5	6 2	13 1	0 0	6 1
INDIAN POINT 2 and 3, Buchanan, N Y (Unıt 2, Consolıdated Edıson Co of New York, Unıt 3, Power Authorıty of State of N Y)	50-247 50-286	PWR (West) PWR (West)	2758 2760	873 965	8/74 4/76	0 0 65 1	0 0 56 1	0 0 102 2	60 0 54 4	0 0 35 7	0 0 0 0	0 0 0 0	7 4 15 7
KEWAUNEE, Carlton, Wıs (Wısconsın Publıc Servıce Corp)	50 305	PWR (West)	1650	535	6/74	24 0	58 3	103 8	81 5	0 0	0 0	0 0	2 4
LA SALLE 1 and 2, Seneca, Ill (Commonwealth Edıson Co)	50-373 50 374	BWR (GE) BWR (GE)	3323 3323	1078 1078	1/84 10/84	0 0 105 3	49 9 103 3	97 7 101 1	56 8 62 6	0 0 0 0	11 6 0 0	0 0 0 0	8 1 13 5
LIMERICK 1 and 2, Pottstown, Pa (Phıladelphıa Electrıc Co)	50 352 50-353	BWR (GE) BWR (GE)	3293 3293	1055 1055	2/86 1/90	96 4 64 3	93 8 0 0	55 6 72 4	67 2 71 4	0 0 4 1	0 0 0 0	37 0 0 0	4 3 8 3
MAINE YANKEE, Lıncoln County, Maıne (Maıne Yankee Atomıc Power Co)	50 309	PWR (CE)	2560	790	12/72	105 6	0 0	103 0	71 4	0 0	99 8	0 0	7 6
McGUIRE 1 and 2, Cowans Ford Dam, N C (Duke Power Co)	50 369 50-370	PWR (West) PWR (West)	3411 3411	1180 1180	12/81 3/84	101 4 102 3	51 4 101 0	99 3 97 0	60 6 71 6	0 0 0 0	44 0 0 0	0 0 0 0	12 8 8 3
MILLSTONE POINT 1, 2, and 3, Waterford, Conn (Northeast Nuclear Energy Co)	50 245 50 336 50 423	BWR (GE) PWR (CE) PWR (West)	2011 2560 3411	660 870 1150	3/71 12/75 4/86	69 7 100 7 0 0	0 0 39 1 99 9	0 0 0 0 77 3	71 3 66 1 72 9	20 7 0 0 0 0	0 0 57 0 0 0	0 0 100 0 19 1	10 0 14 1 10 0
MONTICELLO, Montıcello, Mınn (Northern States Power Co)	50 263	BWR (GE)	1670	545	6/71	78 0	0 0	86 8	71 7	0 0	0 0	10 5	4 0
NINE MILE POINT 1 and 2, Oswego, N Y (Nıagara Mohawk Power Corp)	50 220 50-410	BWR (GE) BWR (GE)	1850 3323	620 1080	12/69 3/88	22 0 91 9	89 6 97 3	91 4 97 7	54 6 46 7	0 0 5 7	0 0 0 0	0 0 0 0	25 4 23 0
NORTH ANNA 1 and 2, Louısa County, Va (Vırgınıa Electrıc & Power Co)	50-338 50 339	PWR (West) PWR (West)	2775 2775	907 907	6/78 12/80	64 6 100 8	45 6 99 1	99 5 99 3	64 5 75 5	0 0 0 0	44 4 0 0	0 0 0 0	12 5 6 0
OCONEE 1, 2, and 3, Oconee County, S C (Duke Power Co)	50 269 50 270 50-287	PWR (B&W) PWR (B&W) PWR (B&W)	2568 2568 2568	887 887 887	7/73 9/74 12/74	100 6 101 6 1 3	93 8 101 5 99 0	99 6 101 4 97 3	69 9 70 2 71 0	0 0 0 0 0 0	4 4 0 0 0 0	0 0 0 0 2 1	11 2 9 9 10 8

(Table contınues on the next page)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		April	May	June	Cumulative (lifetime) thru 6/30/91	April	May	June	Cumulative (lifetime) thru 6/30/91
OYSTER CREEK, Oyster Creek, N J (Jersey Central Power & Light Co)	50-219	BWR (GE)	1930	650	12/69	0 0	0 0	2 8	54 4	0 0	0 0	0 0	11 5
PALISADES, Covert Township, Mich (Consumers Power Co)	50-255	PWR (CE)	2200	805	12/71	24 0	107 0	106 5	48 7	10 4	0 0	0 0	32 2
PALO VERDE 1, 2, and 3, Wintersburg, Ariz (Arizona Public Service Co)	50-528 50 529 50-530	PWR (CE) PWR (CE) PWR (CE)	3817 3817 3817	1270 1270 1270	2/86 9/86 1/88	102 8 101 7 37 5	102 2 102 4 0 0	101 8 102 2 67 9	49 7 66 6 63 7	0 0 0 0 0 0	0 0 0 0 0 0	0 0 0 0 13 3	22 3 7 2 8 2
PEACH BOTTOM 2 and 3, York County, Pa (Philadelphia Electric Co)	50-277 50-278	BWR (GE) BWR (GE)	3293 3293	1065 1065	7/74 12/74	0 0 99 9	33 9 9 5	61 4 93 5	50 1 52 7	0 0 0 0	53 7 84 8	32 5 0 0	14 7 12 7
PERRY 1, Perry, Ohio (Cleveland Electric Illuminating Co)	50-440	BWR (GE)	3579	1205	11/87	98 9	99 2	99 4	67 1	0 0	0 0	0 0	8 2
PILGRIM 1, Plymouth, Mass (Boston Edison Co)	50 293	BWR (GE)	1998	655	12/72	84 6	0 0	0 0	47 1	0 0	100 0	0 0	12 5
POINT BEACH 1 and 2, Manitowoc County, Wis (Wisconsin–Michigan Power Co , Wisconsin Electric Power Co)	50-266 50-301	PWR (West) PWR (West)	1518 1518	497 497	12/70 10/72	102 2 101 7	27 0 102 9	97 5 102 1	74 4 81 8	0 0 0 0	5 1 0 0	1 6 0 0	1 0 1 1
PRAIRIE ISLAND 1 and 2, Red Wing, Minn (Northern States Power Co)	50-282 50-306	PWR (West) PWR (West)	1650 1650	530 530	12/73 12/74	100 8 101 8	75 5 101 6	0 0 100 1	81 0 85 0	0 0 0 0	0 0 0 0	0 0 0 0	5 5 3 1
QUAD CITIES 1 and 2, Rock Island, Ill (Commonwealth Edison Co)	50-254 50-265	BWR (GE) BWR (GE)	2511 2511	789 789	2/73 3/73	0 0 98 0	59 5 93 8	14 5 92 8	64 5 64 4	0 0 0 0	28 8 0 0	83 5 0 0	5 8 8 0
RANCHO SECO, Sacramento County, Calif (Sacramento Municipal Utility District)	50-312	PWR (B&W)	2772	918	4/75	0 0	0 0	0 0	34 2	0 0	0 0	0 0	42 7
RIVER BEND 1, St Francisville, La (Gulf States Utilities Co)	50-458	BWR (GE)	2894	934	6/86	72 1	94 2	93 6	68 6	22 6	0 0	0 0	7 7
ROBINSON 2, Hartsville, S C (Carolina Power & Light Co)	50 261	PWR (West)	2200	700	3/71	66 8	103 5	106 7	63 1	0 0	0 0	0 0	15 3
SALEM 1 and 2, Salem, N J (Public Service Electric & Gas Co)	50-272 50-311	PWR (West) PWR (West)	3423 3423	1090 1115	6/77 10/81	0 0 94 4	98 9 60 3	69 7 95 5	56 3 56 9	0 0 0 0	0 0 0 0	27 0 0 0	22 2 22 9

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power MW(t)	Design power MW(e)	Commercial operation date	MDC unit capacity, % April	MDC unit capacity, % May	MDC unit capacity, % June	MDC unit capacity, % Cumulative (lifetime) thru 6/30/91	Forced outage rate, % April	Forced outage rate, % May	Forced outage rate, % June	Forced outage rate, % Cumulative (lifetime) thru 6/30/91
SAN ONOFRE 1, 2, and 3, Camp Pendleton, Calif (Southern California Edison Co)	50 206 50-361 50-362	PWR (West) PWR (CE) PWR (CE)	1347 3410 3410	436 1070 1080	1/68 8/83 1/84	6 1 68 9 89 0	21 1 61 1 49 5	68 7 101 2 64 8	50 6 69 9 69 3	51 4 9 0 10 1	15 8 35 4 47 6	0 0 0 0 32 9	19 4 7 2 7 8
SEABROOK 1, Seabrook, N H (Public Service Co of New Hampshire)	50-443	PWR (West)	3411	1150	8/90	92 6	99 5	80 6	85 2	4 9	0 0	11 1	9 9
SEQUOYAH 1 and 2, Daisy, Tenn (Tennessee Valley Authority)	50 327 50-328	PWR (West) PWR (West)	3423 3423	1148 1148	7/81 6/82	98 9 100 2	100 2 99 2	99 0 96 2	46 7 49 7	0 0 0 0	0 0 0 0	0 0 0 0	42 9 38 0
SHEARON HARRIS 1, Bonsal, N C (Carolina Power & Light Co)	50 400	PWR (West)	2775	900	5/87	47 4	13 2	87 7	72 7	0 0	0 0	7 8	4 5
SOUTH TEXAS 1 and 2, Bay City, Tex (Houston Lighting and Power Co)	50-498 50 499	PWR (West) PWR (West)	3800 3800	1250 1250	8/88 6/89	0 0 82 1	99 2 88 6	95 8 98 6	55 7 64 6	0 0 15 1	0 0 9 0	0 0 0 0	15 8 16 9
ST LUCIE 1 and 2, Hutchinsons Island, Fla (Florida Power & Light Co)	50 335 50-389	PWR (CE) PWR (CE)	2560 2560	830 830	12/76 6/83	103 5 103 4	90 9 98 1	92 8 96 5	75 3 84 2	0 0 0 0	1 9 0 0	0 0 0 0	4 2 5 5
SUMMER 1, Broad River, S C (South Carolina Electric & Gas Co)	50-395	PWR (West)	2775	900	1/84	77 0	74 3	99 7	70 7	0 0	0 0	0 0	7 1
SURRY 1 and 2, Surry County, Va (Virginia Electric & Power Co)	50 280 50-281	PWR (West) PWR (West)	2441 2441	788 788	12/72 5/73	98 8 76 1	98 2 0 0	96 5 2 1	58 1 57 9	0 0 0 0	0 0 0 0	0 0 87 5	19 9 15 3
SUSQUEHANNA 1 and 2, Berwick, Pa (Pennsylvania Power & Light Co)	50-387 50 388	BWR (GE) BWR (GE)	3293 3293	1065 1065	6/83 2/85	96 9 22 7	99 9 65 6	100 2 98 8	72 1 76 9	0 0 0 0	0 0 0 0	0 0 0 0	8 1 5 9
THREE MILE ISLAND 1, Three Mile Island, Pa (Metropolitan Edison Co)	50-289	PWR (B&W)	2772	906	12/73	93 6	90 9	89 0	48 1	0 0	0 0	0 0	45 5
TROJAN, Columbia, Oreg (Portland General Electric Co)	50-344	PWR (West)	3411	1130	5/76	8 3	0 0	0 0	56 6	87 7	0 0	0 0	13 3
TURKEY POINT 3 and 4, Dade County, Fla (Florida Power & Light Co)	50 250 50-251	PWR (West) PWR (West)	2200 2200	693 693	12/72 9/73	0 0 0 0	0 0 0 0	0 0 0 0	61 0 60 7	0 0 0 0	0 0 0 0	0 0 0 0	12 5 12 0

(Table continues on the next page)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		April	May	June	Cumulative (lifetime) thru 6/30/91	April	May	June	Cumulative (lifetime) thru 6/30/91
VERMONT YANKEE, Vernon, Vt (Vermont Yankee Nuclear Power Corp)	50-271	BWR (GE)	1593	514	11/72	85 5	99 5	79 8	73 1	15 0	0 0	17 1	5 6
VOGTLE 1 and 2, Waynesboro, Ga (Georgia Power Co)	50-424 50-425	PWR (West) PWR (West)	3411 3411	1157 1157	6/87 5/89	98 1 72 1	99 7 89 3	99 3 100 4	81 4 81 0	0 0 0 0	0 0 7 8	0 0 0 0	7 7 2 8
WASHINGTON NP 2, Richland, Wash (Washington Public Power Supply System)	50-397	BWR (GE)	3323	1100	12/84	93 5	0 0	0 0	58 2	0 0	0 0	100 0	8 8
WATERFORD 3, Taft, La (Louisiana Power & Light)	50 382	PWR (CE)	3410	1104	9/85	46 6	3 6	92 8	77 1	0 0	19 6	5 1	4 6
WOLF CREEK 1, Burlington, Kans (Kansas City Power & Light Co)	50-482	PWR (West)	3411	1170	9/85	76 7	63 1	98 1	77 0	0 0	0 0	0 0	3 7
YANKEE ROWE, Rowe, Mass (Yankee Atomic Electric Co)	50-29	PWR (West)	600	175	11/60	83 3	95 6	59 5	74 6	3 2	0 0	29 7	4 9
ZION 1 and 2, Zion, Ill (Commonwealth Edison Co)	50-295 50-304	PWR (West) PWR (West)	3250 3250	1040 1040	12/73 9/74	0 0 63 7	49 6 0 0	77 4 49 3	56 5 60 6	100 1 33 7	43 2 100 0	7 0 42 6	16 9 15 3

[a]The information in this table is obtained from NRC Publication NUREG-0020, Vol 15, Nos 5, 6, and 7

Table 3 Power Generation During the Second Quarter of 1991

Power generation	1989	1990	April	May	June	Year-to-date
Gross electrical, MW(e)h	555 666 518	605 169 082	43 758 938	49 243 365	56 879 375	308 674 132
Net electrical, MW(e)h	528 204 992	575 991 274	41 625 761	46 790 893	54 183 382	294 036 467
Average unit factors, %						
Service	68 2	71 1	61 2	67 1	77 8	70 9
Availability	68 5	71 1	61 2	67 1	77 8	70 9
Capacity						
MDC	63 3	67 0	58 3	62 8	73 6	67 5
DER	61 9	65 5	57 0	61 4	72 0	66 0
Forced outage rate	11 2	9 7	11 1	13 0	11 1	11 5

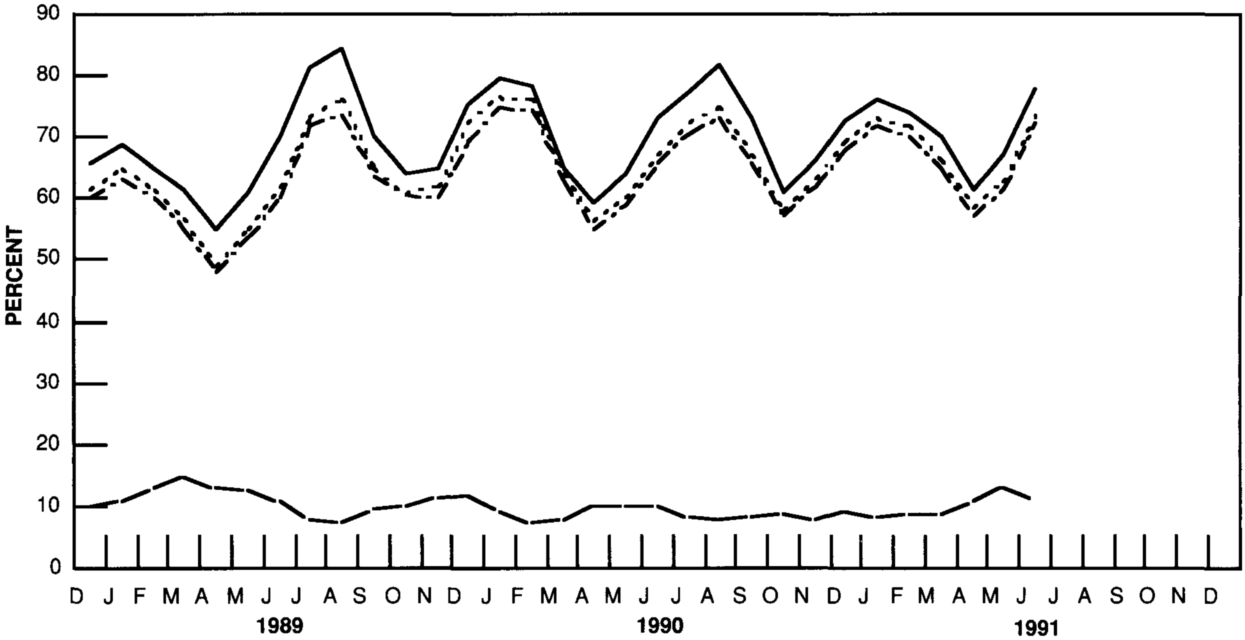

Fig. 1 Average unit availability, capacity factors, and forced outage rate. ——, availability factor. ...., MDC capacity factor. . — . —, DER capacity factor. — — —, forced outage rate. Data through February 1990 are obtained from NUREG-0200; data for the remainder of 1990 were obtained from the NRC Office of Information Resources Management. 1991 data are obtained from the magnetic-media version of NUREG-0200.

of this journal. The reader's attention is also called to the regular features "General Administrative Activities," which deals with more general aspects of regulatory and legal matters, "Waste and Spent Fuel Management," which covers legislative, administrative, and technical matters related to the back end of the fuel cycle and to management of radioactive wastes in general.

TVA RESTARTS BROWNS FERRY UNIT 2

The Tennessee Valley Authority (TVA) restarted Unit 2 of the Browns Ferry Nuclear Plant (BFNP) on May 23, 1991, completing an extensive recovery effort for an operating plant.[1] The BFNP consists of three BWR electric generating units, each rated at

1098 MW(e). TVA shut down Unit 2 for refueling in September 1984 and shut down Units 1 and 3 in March 1985 because of NRC concerns regarding declining performance at BFNP. All three units remained shut down until Unit 2 was restarted.

In the course of its restart efforts, TVA was planning to start up and shut down the reactor several times as part of a program designed to bring the plant to full-power production in 90 days. During this time, personnel were to perform more than 30 tests to verify that all plant systems function as designed. According to O. J. Zeringue, vice president of Browns Ferry operations, the tests program was to be stopped at specific points in the test sequence to evaluate the plant and assure a controlled return to full-power operation.[1]

TVA said its work at Browns Ferry Unit 2 included a complete review and verification of plant design, physical improvements in the equipment and systems, increased management involvement at all levels, increased standards for training and performance, and multiple reviews to ensure readiness. During the seven years the reactor was idle, TVA devoted nearly 15 million engineering working hours to establish the engineering criteria, designs, and documentation for Unit 2 startup and operation and more than nine million working hours to make plant improvements identified by TVA and NRC, according to Zeringue.

Not everyone was pleased about the fact that Browns Ferry 2 was to be operated again, however. The anti-nuclear advocacy group Public Citizen denounced the restart, stating that the restart of the reactor would "likely come back to haunt both TVA and NRC."[1] According to Public Citizen, the restart of Browns Ferry could not be justified on safety grounds and is ill-advised on economic grounds. "Notwithstanding its expenditure of hundreds of millions of dollars to restart Browns Ferry, it is likely that the TVA will eventually be forced to admit that it would have been more economic to have permanently closed the plant in favor of safer, cheaper, and cleaner alternatives," Public Citizen asserted.

NRC STAFF VALIDATES EXISTING SEISMIC DESIGN OF DIABLO CANYON

After reevaluating the seismic design of Pacific Gas and Electric Company's (PG&E) Diablo Canyon Nuclear Power Plant, the NRC has concluded that the existing seismic design continues to provide an adequate margin of safety.[2] This conclusion was still subject to the completion of certain confirmatory calculations by PG&E.

At the time construction permits for the Diablo Canyon facilities were issued (in April 1968 for Unit 1 and in December 1970 for Unit 2), the approved seismic design bases were based on two hypothetical events—a magnitude 7.25 earthquake on the Nacimiento fault and a magnitude 6.75 aftershock at the site associated with a large earthquake on the San Andreas fault. At the time, it was believed that there was no other fault capable of causing more severe ground motions at the site. In 1971, however, information became available on the existence of the Hosgri fault, which is located about three miles offshore from the site. As the result of detailed investigations of this fault, and prior to issuance of the operating licenses, the plant seismic design was reevaluated and the plant was upgraded so that it would safely withstand the ground motions that would result from a postulated earthquake of magnitude 7.5 on the Hosgri fault at its closest approach to the site. Full-power operating licenses for Diablo Canyon Units 1 and 2 were issued in 1984 and 1985, respectively.

During its review of the application for operating licenses for the Diablo Canyon plant in 1978, the NRC's Advisory Committee on Reactor Safeguards (ACRS) recommended that the seismic design be reevaluated about ten years later (i.e. in about 1988). The NRC therefore placed conditions on operating license in accordance with this ACRS recommendation. Specifically, the NRC required that a reevaluation program consisting of the following four elements be undertaken:

1. Identification, examination, and evaluation of all relevant geologic and seismic data, information, and interpretations that would become available after 1979 to update the information available concerning the geology, seismology, and tectonics in the region of Diablo Canyon and, if needed, reevaluation of earlier information and acquisition of additional new data.
2. Reevaluation of the magnitude of the earthquake used to determine the seismic bases of Diablo Canyon with information from element one.
3. Reevaluation of the ground motions at the site on the basis of the results from element two with full consideration of site and other relevant effects.
4. An assessment of the significance of conclusions drawn from elements one, two, and three—with the use of probabilistic risk analysis and deterministic studies as necessary to assure adequacy of seismic margins.

The PG&E addressed all four elements as part of its Long-Term Seismic Program (LTSP). The NRC conducted an in-depth review of PG&E's response to each of these elements and concluded that:

1. The geological, seismological, and geophysical investigations and analyses were the most extensive, thorough, and complete ever conducted for a nuclear facility in this country and have advanced the state of knowledge in these disciplines significantly.
2. The Hosgri fault is the seismic source that could cause the maximum vibratory ground motion at the Diablo Canyon site; the maximum earthquake that could occur on that fault would have a magnitude of 7.2 and its epicenter could be as close as about 4.5 km from the site.
3. The NRC's own estimate of vibratory ground motion at the Diablo Canyon site is equal to or less than PG&E's estimates over part of the frequency range of interest but exceeds PG&E's estimates over another part of the range.
4. The PG&E concluded that plant seismic margins were adequate to withstand the NRC staff's vibratory ground-motion estimates in element three, and the NRC found PG&E's conclusion to be acceptable but will nevertheless require the utility to perform calculations to confirm its conclusion. As a result of a separate reevaluation by the NRC, PG&E plans to modify all safety-related masonry walls.

On the basis of the preceding conclusions, the NRC reported that PG&E had satisfied the license condition. Finally, the NRC staff concluded that, for future plant design modifications, its own higher ground-motion estimates from element three given previously should be used to provide further assurance that plant seismic design margins remain acceptable.

NRC CONSIDERING TMI 2 LICENSE CHANGE TO ALLOW ONLY POSSESSION OF REACTOR AND FUEL

The NRC considered a request from General Public Utilities (GPU) Nuclear Corporation to change the operating license for Three Mile Island 2 (TMI 2) to a possession-only license and to allow for long-term storage of the facility.[3] A possession-only license authorizes a licensee to possess both the reactor and reactor fuel but not to operate the reactor.

Since the Mar. 28, 1979, accident, GPU Nuclear has been conducting a long-term cleanup and defueling effort at TMI 2. The licensee determined that placing TMI 2 in long-term storage would result in radiation dose reductions for personnel by postponing further decontamination efforts that might also adversely impact operations at TMI 1. GPU Nuclear proposed to place TMI 2 in long-term storage until the operating license for TMI 1 expires on Apr. 19, 2014. At that time the licensee would begin decommissioning both TMI units for ultimate release of the facility for unrestricted access. The question of whether a license extension for the operation of TMI 1 past April 2014 might be sought and what such an extension might mean for the schedule for the decommissioning of TMI 2 has not yet been addressed.

NRC TO CONSIDER RANCHO SECO POSSESSION-ONLY LICENSE REQUEST

The NRC's Atomic Safety and Licensing Board (ASLB) was considering an application from Sacramento Municipal Utility District (SMUD) for a possession-only license for its Rancho Seco nuclear generating station.[4] SMUD decided to cease operations permanently at Rancho Seco after residents voted in June 1988 that SMUD not be permitted to operate the plant.

A public-interest group, the Environmental Conservation Organization (ECO), filed a petition to intervene and a request for a hearing on the SMUD application. The ASLB was planning to hold a pre-hearing conference to permit representatives of SMUD and the NRC to respond to ECO's arguments concerning its request for a hearing and to consider ECO's proposed contentions related to the SMUD application.

COMMISSION APPROVES LICENSE TO DISMANTLE SHOREHAM

The dramatic controversy surrounding the $5.5 billion construction of the Shoreham Nuclear Power Station in Long Island, N.Y., recently came another step closer to drawing to a close.[5] In mid-June, 1991, the NRC officially denied a petition sought by two proponents of opening the nuclear plant, the Shoreham-Wading River Central School District and the Scientists and Engineers for Secure Energy, that attempted to thwart NRC's authority to issue a "possession only" for Shoreham.

The decision by NRC to go ahead with issuance of the license means that the Long Island Lighting Company, which owns Shoreham, may possess, but not operate, the nuclear plant and paves the way for transfer of ownership to the state to be followed by the removal of all nuclear equipment. Once the equipment is dismantled, the state will study the possibility of converting Shoreham to a gas-fired plant. The projected cost of stripping down the plant, to be borne by LILCO ratepayers, is $186 million.

The petitioners requested the Commission to (1) refrain from issuing a "possession only" license, (2) stay further proceedings by the ASLB, and (3) stay further NRC staff review of other pending applications for related amendments to the Shoreham license while awaiting the outcome of pending litigation before the New York Court of Appeals regarding the Shoreham facility. The crux of the petitioners' argument was that the National Environmental Policy Act (NEPA) requires NRC to publish an Environmental Impact Statement considering "resumed operation" of Shoreham as an alternative to decommissioning. Further, the petitioners argued that this NEPA duty must be carried out immediately because of preliminary activities by NRC staff. The regulatory body denied the petitioners' appeal, disagreeing with their basic NEPA argument.

The only electricity ever generated at Shoreham was the result of low-level testing. Under pressure from Gov. M. M. Cuomo (D-N.Y.), who cited inadequate evacuation procedures for those on the far tip of Long Island as the basis of his argument, the plant was never allowed to begin commercial operation.

Cuomo persuaded LILCO in 1989 to sell the plant to the state for one dollar in exchange for ten years of rate increases. After purchasing the plant, the state planned to take it apart. But the Administration intervened, and, as part of its attempt to resurrect nuclear energy, made Shoreham a central issue in its campaign. Secretary of Energy J. D. Watkins vowed in April 1989 to do everything he could to keep Shoreham alive.

The Commission's ruling puts a serious damper on the hopes and plans of the Administration and the Department of Energy to reopen Shoreham. Supporters of restarting Shoreham can and undoubtedly will appeal the ruling in the Federal Court of Appeals, which has jurisdiction over the Commission's decisions.

In a separate but related court action, LILCO settled its $400 million suit against the General Electric Company (GE) out of court. LILCO had held GE liable for the cost of major repairs performed on the containment vessel at the Shoreham plant. The agreement calls for GE to provide LILCO with "certain goods and services at reduced costs" and to assist with removal of nuclear fuel from the plant.

In 1974, GE reported that it had discovered several defects in its containment systems that could possibly damage the structure of vessel shells of Mark I and II type plants if an accident occurred. LILCO and several other utilities maintained that GE knew of the flaws before selling the product. GE said the problems were related to new requirements set by NRC after the plants were built.

SEVEN NEW FINES DURING REPORTING PERIOD

Seven civil penalty fines have been levied by the NRC on licensees during the three-month period covered by this report (April, May, and June 1991). In each case the affected licensee must report to the NRC on the causes and proposed corrections of the problem or violation that led to the imposition of the fine. The licensee also has 30 days from the date of notification to either pay the penalty or protest its imposition in whole or in part. Each of these seven cases is briefly described here.

Hatch: Potential for a Radiation Overexposure to Personnel

The NRC staff proposed a $50 000 civil penalty against Georgia Power Company, the licensee for the Hatch nuclear power plant, for alleged violation of NRC requirements.[6]

The NRC said the action was being taken following a review by the agency of radiation protection activities associated with a traversing incore probe event on Feb. 11, 1991, which the NRC inspectors said created a substantial potential for a radiation overexposure to certain plant personnel.

The NRC notified the company in a letter dated Apr. 15, 1991, that the problem occurred during troubleshooting activities involving a computer system at the plant. Two plant employees and a contract employee were working to resolve the computer problem when they decided to break for lunch and informed a Unit 1 control room operator that they were finished with the morning's procedure. Later the health physics office contacted the control room for permission to enter the room where the probe was located to perform a decontamination survey. The Health Physics staff was told that no one would be operating the system during the time the Health Physics personnel would be doing their survey; however, a health physics technician preparing to enter the room a short time later noticed that a survey meter was indicating a higher than allowable radiation reading before he entered the area. When the probe operating panels were checked, it was found that the contract employee had returned and had withdrawn a probe into a shield and was operating another in an attempt to solve the computer problem without the knowledge of the control room operators. The situation was corrected immediately, and no overexposure occurred.

However, the NRC said it was concerned because the event created a substantial potential for a radiation overexposure to personnel and indicated a serious failure to control manipulations in the control room.

Dresden 2: Potential Leakage Path from Containment

The NRC staff proposed a $100 000 fine against Commonwealth Edison Company, operator of the Dresden nuclear power plant, because of the existence of a potential leakage path from the reactor containment at Dresden Unit 2 (Ref. 7). The violation did not result in the release of radioactivity, but the plant's ability to contain radioactive gases would have been reduced in the unlikely event of a nuclear plant accident.

During testing of the containment in December 1990, the utility discovered excessive leakage through a valve flange. The testing showed that the leakage rate exceeded NRC limits at the pressures that might occur during an accident. Unit 2 was shut down for refueling at the time of the test.

The leakage was caused by improper maintenance work on the valve in February 1989 and the utility's failure to test the valve flange for leakage after the maintenance was completed. The valve is in piping that can be used to vent air and gases from a portion of the containment.

When the problem was discovered, Commonwealth Edison promptly corrected the problem. The company also revised its valve maintenance and testing procedures to avoid similar problems in the future.

Sequoyah: Operators Fail to Respond to Alarm in Control Room

The NRC staff has imposed a $75 000 fine on TVA for an alleged failure to respond to an alarm indicator in the control room at the Sequoyah nuclear power plant.[8] The action stems from an event at Sequoyah on Jan. 24, 1991, when plant personnel discovered that an air start accumulator on an emergency diesel generator was depressurized to a low level and that plant operators had not responded to an alarm indicator in the control room, which indicated the problem.

There was no actual emergency associated with the event, but the NRC said that the condition existed for up to two hours and was not recognized until a worker responded to an audible alarm in the diesel building and notified the control room of the problem. The base civil penalty for this type of violation is $50 000, but this was increased to $75 000 in the present instance because of poor past performance associated with command and control on the part of the plant operations staff.

Yankee Atomic: Quality Control Violations

The NRC staff has proposed a $50 000 fine against the Yankee Atomic Electric Company for alleged violations that were found after one of the emergency diesel generators at the Yankee Nuclear Power Station in Rowe, Mass., failed to start during a routine surveillance test in January 1991 (Ref. 9). The fine was assessed because of violations associated with (1) the failure to ensure adequate training of contractor craft personnel who installed electrical connections during the emergency diesel generator (EDG) changeout in the 1990 refueling outage and (2) the failure to establish and implement an effective Quality Control inspection program.[10] The violations were categorized as a Severity Level III problem because the significant lack of contractor oversight resulted in defects that affected the operability of a safety system. The violations resulted in over 90 quality deficiencies in a number of safety systems.

In a letter to Yankee Atomic, T. T. Martin, NRC Region I Administrator, said "the NRC considers these violations significant because the deficient electrical connections had the potential to be a common mode failure since the work was performed by the same contractor on each EDG."

Brunswick: Continuing Human Performance Problem

The NRC staff informed Carolina Power and Light Company (CP&L) that it was proposing to fine the company $87 500 for three violations of NRC requirements at the Brunswick Steam Electric Power Plant.[11] The action was based on three violations classified in the aggregate as a Severity Level III problem as the result of "the recurring personnel performance problems that collectively represent a potentially significant lack of attention or carelessness toward licensed responsibilities."[12] The violations specifically involved (1) the failure of instrument and control technicians to perform and document steps in a calibration procedure; (2) performance of work on the No. 1 EDG without a written procedure, which resulted in substantial damage to the camshaft; and (3) the failure of two auxiliary operators to deenergize the correct d-c control power beaker for a tag-out. The base civil penalty was mitigated by 25% because of the corrective actions undertaken by the licensee. However, a 100% escalation factor was applied for the licensee's poor past performance.

In his letter informing CP&L of the fine, S. D. Ebneter, NRC regional administrator in Atlanta, said these violations—although individually of low safety impact—"represent a continuing human performance problem that is of serious concern to the NRC." He noted that such problems at Brunswick have required NRC enforcement action several times during the past year.

Ebneter said NRC recognizes that CP&L already has obtained "tangible results" in some areas from improvement actions it has initiated, and he added that NRC realizes that persistent human performance problems take time to correct. But he said NRC remained concerned that Brunswick management had not yet been able to correct the root cause of recurring personnel performance problems.

River Bend: Violation of Radiation Protection Requirements

The NRC staff proposed a fine against Gulf States Utilities in the amount of $37 500 for violations of NRC radiation protection requirements at the River Bend Station.[13] NRC is citing Gulf States for incidents that occurred on Apr. 1 and 28, 1991, when barriers to a high radiation area had not been maintained, and on Apr. 24, 1991, when inspectors observed at least three workers entering radiation areas without reviewing radiation survey documents to become familiar with radiological conditions there.

In his letter informing Gulf States of the civil penalty, NRC regional administrator Martin noted that River Bend has had a history of such violations documented in NRC inspection reports as early as March 1989.

These earlier violations, Martin said, involved the failure to maintain the security of very high radiation areas, as well as a failure by employees either to have the required radiation monitoring equipment or to be escorted by a radiation protection technician upon entering a high radiation area. On the basis of the April 1991 violations, Martin added, it appears that such problems continue to exist despite Gulf States' efforts to prevent them.

"Although none of these violations has led to radiation exposures that exceeded NRC limits for radiation workers," Martin wrote, "violations of this type create the potential for significant exposures to personnel. . . ." He added that the violations also "collectively indicate a lack of attention toward licensed responsibilities on the part of . . . personnel who have been trained to understand these requirements."

Martin said a Gulf States task force formed in December 1990 has begun to correct the situation. Its actions have included enhancements of radiation area boundaries, stepped-up employee and supervisor training, increased checks of all radiation area barricades, and disciplinary actions against both employees and supervisors.

Gulf States' discovery and prompt reporting of some of the violations, along with its extensive corrective measures, resulted in the civil penalty being reduced from the $50 000 base amount for violations of this sort.

St. Lucie: Inadequate Verification of Valve Position

The NRC staff has proposed a $37 500 civil penalty against Florida Power and Light Company (FP&L) for alleged violation of NRC requirements at the St. Lucie nuclear power plant.[14] NRC officials said the action was being proposed because of a condition identified by the plant staff on Apr. 26, 1991, which indicated that one of two containment spray systems in Unit 2 had been inoperable since about Nov. 29, 1990, because of an error in positioning a valve. During that time period the other containment spray system was unavailable or out of service for relatively short time periods. In the course of some postulated accident scenarios, the containment spray system would be needed to reduce temperature and pressure inside the containment building.

The NRC said plant rules call for personnel to provide a weekly verification of the correct position of locked valves by unlocking them and physically verifying valve position manually in the closing direction. The valve in question is in a horizontal line about 12 ft above the floor in a pipe tunnel, and workers apparently relied upon visual verification by observing a broken position indicator that inaccurately indicated that the valve was open

The NRC officials said the company took prompt action to correct the situation when it was identified and has taken steps to assure that plant personnel properly perform these functions in the future.

REFERENCES

1 *At Energy Clearing House* 37(22) 1 (May 31, 1991)
2 *At Energy Clearing House* 37(25) 8 (June 21, 1991)
3 NRC Press Release 91 44, Apr 29, 1991
4 NRC Press Release 91-61, June 14, 1991
5 *At Energy Clearing House* 37(24) 1 (June 14, 1991)
6 NRC Press Release 91-40, Apr 16, 1991
7 NRC Press Release 91 41, Apr 22, 1991
8 NRC Press Release 91 50, May 14, 1991
9 NRC Press Release 91 56, May 31, 1991
10 *At Energy Clearing House* 37(24) 8 (June 14, 1991)
11 NRC Press Release 91-57, June 3, 1991
12 *At Energy Clearing House* 37(24) 8 (June 14, 1991)
13 NRC Press Release 91 63, June 20, 1991
14 NRC Press Release 91-67, June 26, 1991

Recent Developments

Edited by E. G. Silver

General Administrative Activities

Compiled by M. D. Muhlheim[a] and E. G. Silver[a]

"General Administrative Activities" summarizes selected current topics that are related to nuclear safety but do not fit elsewhere in the journal. Included in this issue are items reported during April, May, and June 1991. Subjects discussed, among others, are the current status on power-plant license renewal, operator requalification exams, and Part 52 design certification.

ACRS COMMENTS ON SEVERAL ISSUES

The Advisory Committee on Reactor Safeguards (ACRS) has issued a number of letter reports to the NRC during the period covered by this report (April, May, and June 1991). A number of them are briefly described and excerpted here.

Proposed Criteria to Accommodate Severe Accidents in Containment Design

Between May 8 and 11, 1991, the ACRS discussed the development of criteria that would incorporate explicit consideration of severe accidents into requirements for containment design. The report reads, in part, as follows:[1]

> Our purpose in writing this report is to describe and recommend a possible course by which the NRC could develop an improved set of requirements for the design of containment systems for future nuclear power plants. These requirements would include definition of specific challenges posed by severe accidents. They would be promulgated by revisions and additions to 10 CFR Part 50, primarily to Appendix A, "General Design Criteria for Nuclear Power Plants." Implementation also would require new regulatory guides (RGs). More detail about rule changes and regulatory guides is provided in the Appendix.
>
> We intend this to be a description of a general approach that could be taken. Guidelines for the regulatory guides are provided primarily to illustrate that approach. Final detail and quantification should be developed and justified by the staff with input and review by industry and the reactor safety community.
>
> The new requirements would be applicable to future plants, those not yet designed. We would exclude the "evolutionary" LWRs, for which designs are well advanced. We believe the new criteria can and should be adopted for use in the development and licensing of the "passive" plant designs.
>
> An alternative or interim approach would be to adopt the general process we propose as an extension of the "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants" published in August 1985. This could be more easily and rapidly adopted, in comparison with the rulemaking approach, as a guide for designers and staff reviewers, and as a basis for design certification. A disadvantage is that the "policy" approach would be subject to less rigorous reviews and more limited input from the general body of available expertise on severe accidents and containment performance. We recommend the rulemaking approach.
>
> Future licensing responsibilities of the NRC may include nuclear power plants other than LWRs. Our proposal is for application only to LWRs. As discussed above, we propose that new containment requirements be implemented through changes in appropriate sections of the General Design Criteria. The introduction to 10 CFR Part 50, Appendix A (issued in 1971), states that these criteria apply for "water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." There are some general principles that could be applicable to other types of plants. Such application is, however, a task for another day.

[a]Oak Ridge National Laboratory.

There will be debate over our proposal It will center on the question of whether it is better to continue with the present set of requirements, which it might be argued are good enough, or to develop requirements that reflect what has been learned about severe accidents over the past decade A classical conflict between short-term and long-term costs and benefits exists We recommend that development of new containment design criteria proceed along the lines we have proposed We believe that benefits in safer and more efficiently designed plants and in stabilization of an important part of the regulatory process will be substantial We look forward to the opportunity to interact with you and the staff on this important subject

I BACKGROUND

The primary purpose of the containment and its associated systems in an LWR plant is to mitigate the consequences of severe accidents, those which involve fuel melting and an abundant release of fission products Other important purposes of the containment include housing the nuclear steam supply system and protecting it from external threats, shielding the environment from radiation emanating from the reactor system, and mitigating the releases of radioactive substances caused by normal operation or incidents of lesser scope than severe accidents

Although this primary purpose has been recognized from the beginning, and is perhaps obvious, existing NRC requirements do not account for many severe accident phenomena that could challenge a containment's ability to perform its function

In the early 1960s, licensing authorities and the reactor safety community (including the ACRS) recognized that the risk of a severe accident was real, but remote and largely undefined Rather than await the results of what was seen to be a long and difficult research effort to understand more about severe accidents, a decision was made to use a surrogate accident as a design basis for the containment and to move forward with the development of nuclear power That surrogate accident, a sudden large-break LOCA, coupled with the siting criteria in 10 CFR Part 100, has been the basis for LWR containment design ever since

During the 1979 accident at Three Mile Island 2, a containment designed to the surrogate requirements functioned effectively to protect the public On the other hand, severe accident research and risk assessments performed since 1979 indicate that a broad range of high-energy loads and fission product releases, more severe than at Three Mile Island 2, might threaten containment systems There are indications that certain unlikely severe accident challenges could cause containments to fail, and lead to the release of health-threatening quantities of fission products While the predicted risk from those accidents is small, uncertainty in quantification of the risk is large Improvements in the design of containments could reduce both the risk and the uncertainty

II REVIEW OF EXISTING CONTAINMENT REQUIREMENTS

Formal criteria by which acceptable reactor containments were to be designed and built were established by the Atomic Energy Commission in the 1960s and 1970s General Design Criteria for water-cooled nuclear power plants (10 CFR Part 50, Appendix A), promulgated in 1971, included the following requirements relating to containment

—Criterion 16 specifies "an essentially leak-tight barrier" between the reactor systems and the environment as one of "multiple fission product barriers "

—Criteria 38 through 40 require systems to remove decay heat from the containment to negate pressure buildup that would otherwise result

—Criteria 41 through 43 provide for a system to remove fission products from the containment atmosphere to reduce the consequences of ongoing leakage

—Criterion 50 requires that the containment structure be able to accommodate "the calculated pressure and temperature conditions resulting from any loss-of-coolant accident " This is to be accomplished "without exceeding a design leakage rate and with sufficient margin " It states that the margin should reflect consideration of (1) potential energy sources such as energy in steam generators, limited metal–water reaction that might result from degradation but not failure of the ECCS, (2) limited information on accident phenomena, and (3) conservatism in the calculations There is no requirement in GDC 50 to accommodate severe accidents However, this was remedied in part by 10 CFR 50 34(f) for near term operating licenses, 10 CFR 52 47 for standard design certification, and 10 CFR 50 44 for combustible gas control

—Criteria 51 through 57 provide requirements for containment materials, testing, penetrations and isolation

Reactor siting criteria in 10 CFR Part 100, established in 1962, indirectly determine the maximum leakage rate for which the containment is to be designed Section 100 11 establishes dose limits for the whole body and for the thyroid A referenced document, TID-14844, suggests amounts of radioactive material within containment that are to be assumed in calculating hypothetical doses from post-accident containment leakage TID-14844 also suggests a leakage rate of 0 1 percent of the containment volume per day

Additional guidance is provided in two regulatory guides originally issued in 1970 Regulatory Guide 1 3 is for BWRs and Regulatory Guide 1 4 is for PWRs Each specifies the proportions of the elemental, particulate, and organic forms of the radioiodines that are to be assumed in making dispersion and dose calculations These are, respectively, 91 percent, 5 percent, and 4 percent In addition, Regulatory Guide 1 4 permits the assumption that the leakage rate from containment for PWRs is reduced to one-half the value given in technical specifications after the first 24 hours

III WHY NEW CRITERIA ARE NEEDED

A first purpose of new containment requirements will be to reduce the risk and uncertainty by more directly accounting for severe accident threats than is done with present requirements This should be feasible because in 1991 more is known about the nature of severe accident threats than was known in 1971 Our proposal is simply a way of applying this improved knowledge to provide improved containment systems

A second purpose is to clarify what is expected of applicants and to bring greater coherence to the design review and certification processes Many severe accident considerations are now being factored into staff reviews of advanced reactor designs, but the process by which this is done is not well defined

A third purpose is to help ensure that containments will have greater "robustness " A containment cleverly and narrowly designed to mitigate a set of accidents that has been precisely identified may not be able to cope with the unexpected A truly "robust" containment would have improved capability to deal with the unexpected A containment that has been designed with explicit consideration of a more extensive set of challenges is likely to be more robust than one designed with consideration of only a limited set

IV PROPOSED APPROACH TO DEVELOPMENT OF NEW CONTAINMENT DESIGN CRITERIA

We have previously recommended (ACRS report of May 13, 1987 regarding Safety Goal Policy) a conditional 10 percent failure probability for the containment, reflecting our judgment about the need for assurance of containment performance It is worth recalling that our recommendation was meant as a hedge against uncertainty, to preserve the concept of defense in depth—itself a hedge against uncertainty If all calculations were accurate and credible, all that would matter would be that the population of plants meets the Commission's safety goals, and the identification of containment performance as a separate item would be inappropriate It is because quantitative risk estimates are not perfect that defense in depth is a useful philosophy, and that separate containment performance guidelines make sense

The containment performance objective should serve as guidance to the NRC staff in judging whether requirements for containment design properly reflect the intent of the Commission as expressed in the Safety Goal Policy The conditional containment failure probability should not be simply passed on to applicants for plant licenses Instead, we propose a two-step process to establish new requirements

First, the General Design Criteria in 10 CFR Part 50 would be revised to acknowledge that containments should be designed for a range of challenges that can threaten their function during severe accidents Several different challenges or containment loads would be defined, as discussed in Section V of this Appendix For each, the nature of the challenge would be described in general terms, specifics and quantification would be relegated to a regulatory guide Also, for each, a success criterion would be specified In most cases, success would be defined simply as maintenance of the containment function for an appropriate period following the particular challenge In addition to the GDC changes, certain other regulations concerned with containment would be modified

Second, new regulatory guides would be developed to detail acceptable means to implement the design requirements For the severe accident requirements of GDC 50, regulatory guides would address each challenge

The regulatory guides would provide technical definitions of acceptable means of meeting the general design criteria for containment What we have in mind is a relationship between each GDC requirement and its companion regulatory guide similar to the existing relationship between GDC 35, "Emergency Core Cooling," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models " GDC 35 states that a system shall provide "abundant emergency core cooling " Appendix K gives, in reasonably unambiguous language, a technical definition of the leak that must be accommodated and a definition of the terms "abundant" and "cooling "

Our revised GDC 50 would state the requirement that containments must have the capability to accommodate a specific list of challenges without loss of containment function For each challenge, a regulatory guide would define in unambiguous technical terms, first the challenge, and second, what is meant by the term "accommodate "

The technical content of each regulatory guide should provide as complete and unambiguous a basis for containment system design as can be practically developed For example, the criterion for capacity to accommodate hydrogen combustion might state the total amount of hydrogen to be considered, as a percentage of that which could be generated by complete oxidization of cladding in contact with active fuel, and then require a specific analysis for mixing and stratification The regulatory guide would describe acceptable mixing models, based on containment type

An important aspect of what we are proposing is that the NRC will take responsibility for the important technical judgments necessary to transform knowledge from severe accident research and risk assessments into criteria and requirements that can be used by a designer This would not be done in isolation, review and input from the industry and the reactor safety community should be sought as the rule changes and regulatory guides are developed

In the following sections, we propose revisions to the regulations relating to containment design requirements and also provide information on the content of proposed regulatory guides Although we have not attempted to couch the GDC proposals in regulatory language, we believe that the scope of our description is close to the appropriate scope for the rule In contrast, our proposals for regulatory guides are intended to be only the bare bones of what the guides should contain It will be up to the staff to develop quantifications and to provide appropriate justification We will want to interact with the staff as final details are developed

V RECOMMENDED CHANGES AND ADDITIONS TO 10 CFR PART 50 APPENDIX A "GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS"

We recommend that the following General Design Criteria be changed as indicated

Criterion 16—Containment Design

This criterion specifies an "essentially leak-tight barrier ... for as long as postulated accident conditions require." No changes in wording are necessary but implications of the words would be different. A regulatory guide would specify a definition for leak-tight that is consistent with the overall package of containment requirements. Existing regulatory guides suggest a leakage rate of 0.1 percent of containment volume per day. Present information about severe accidents and the role of containment suggest leakage of 1 percent may be more appropriate. In addition, the accident conditions for which such a leakage limit would apply should reflect other requirements, in particular those in the new GDC 50.

Criteria 38-40—Containment Cooling

These requirements would be changed to reflect the demands placed upon containment cooling systems by other new requirements, especially the proposed new GDC 50(f) and 50(g) below.

Criteria 41-43—Containment Atmosphere Cleanup

These requirements would be changed to reflect the demands placed upon containment atmosphere cleanup systems by other new requirements, especially the proposed new GDC 50(f) below.

Criterion 50—Containment Design Basis

This criterion would be extensively expanded to require that containment systems be designed to accommodate a variety of challenges that could be created by severe-accident conditions. We believe that the challenges can be adequately represented by the eight examples discussed below. Each would be defined in a section [(a) through (h)], with a success criterion identified, and with appropriate supporting regulatory guides. These are not meant to be accident scenarios, but are representative phenomenological challenges.

50(a) Loss of Coolant Accident

The containment system would have the capacity to accommodate pressure and temperature conditions resulting from the blowdown of fluid from a large break LOCA, and in the case of PWRs, from a nonconcurrent blowdown of the secondary system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the accident.

50(b) Fuel–Coolant Interaction

The containment would have capacity to accommodate missiles that could be produced by credible steam explosions within the vessel and to accommodate pressure pulses that could be produced by credible steam explosions outside the reactor vessel and within containment. Steam explosions are characterized by the rapid transfer of thermal energy from molten material to water. Where appropriate, the addition of chemical energy to the thermal energy source would be included in performance calculations.

Leakage should not exceed the rate specified in Criterion 16 following the missile impact or the pressure pulse crediting dynamic response of the containment structure.

50(c) Hydrogen Combustion and Detonation

The containment would have capacity to accommodate pressure pulses produced by static or shock loadings resulting from the combustion or detonation of hydrogen produced during severe accidents. Hydrogen sources to be considered are the in-vessel and ex vessel oxidation of core materials, including (1) core degradation from overheating and melting, (2) steam explosions or high pressure melt ejection in the presence of water, and (3) interaction between molten core material and concrete.

Leakage should not exceed the rate specified in Criterion 16 following the pressure pulse, crediting dynamic response of the containment structure.

50(d) Melt Attack on Containment Structure or Pressure Boundary

The containment design would preclude potential for damage to the containment pressure boundary or essential structure by direct contact of molten core material.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the melt attack.

50(e) High Pressure Melt Ejection

The containment system would have the capacity to accommodate rapid increases in static pressure and temperature caused by heating of the containment atmosphere through the direct transfer of thermal and chemical energy from molten core material ejected at high pressure into the containment, unless such ejection is precluded by design of the reactor system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the melt ejection.

50(f) Corium–Concrete Interaction

The containment system would have the capacity to accommodate the following challenges resulting from the thermal decomposition of concrete by molten corium: (1) the degradation of containment cooling and of cleanup capability due to aerosol formation, (2) slow overpressurization resulting from the evolution of noncondensable gases, (3) functional degradation of structural concrete by erosion, including basemat penetration, and (4) combustion of carbon monoxide.

Challenges to the containment should not be sufficient to render inoperable that equipment required for containment cooling or atmospheric cleanup, nor to cause leakage in excess of the rate specified in Criterion 16 or to allow any release through the basemat within an appropriate time of the onset of the corium–concrete interaction sufficient to cause significant contamination of the groundwater.

50(g) Pressurization from Decay Heat

The containment system would have the capability to accommodate the long-term buildup of pressure resulting from decay heat. This could include an appropriate containment venting system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the accident.

50(h) Elevated Temperatures

Contaınment penetratıons, equıpment necessary for accıdent management, essentıal ınstrumentatıon, and key structural components would have the capacıty to accommodate exposure to elevated contaınment temperatures

Exposure of the noted systems and components followıng exposure to elevated temperatures should not be suffıcıent to cause leakage ın excess of the rate specıfied ın Crıterıon 16 or damage suffıcıent to render ınoperable that equıpment necessary for accıdent management for an approprıate perıod followıng the exposure

Crıterıa 51 53

No changes ın these crıterıa are proposed

Crıterıa 54 57

These would be revısed to be consıstent wıth new Crıterıon 58 Sımplıficatıon of Crıterıa 54-57 may be possıble

In addıtıon to the revısıons to exıstıng crıterıa, descrıbed above, we recommend the followıng new crıterıa as addıtıons to Appendıx A

Crıterıon 58—Provısıon For On Lıne Monıtorıng of Contaınment Isolatıon Status

Thıs new crıterıon would be ıntended to reduce the lıkelıhood of loss of contaınment functıon by contınuous onlıne monıtorıng It must be consıstent wıth Crıterıon 16

Crıterıon 59—Role of Contaınment Structure ın Protectıng Nuclear Components Agaınst External Threats

Thıs new crıterıon would be ıntended to protect the nuclear steam supply system and other essentıal components agaınst credıble aırcraft crashes, explosıons, and other nonnatural threats external to the plant Alternatıvely, the exıstıng Crıterıon 2, whıch calls for resıstance to extreme natural condıtıons, could be revısed to ınclude such threats

Crıterıon 59-A—Assurance of Contaınment Integrıty Durıng Shutdown

Thıs new crıterıon would requıre that contaınments wıll be desıgned to provıde for ease of emergency closure durıng shutdown operatıon ıncludıng statıon blackout condıtıons

VI RECOMMENDED CHANGES TO OTHER REGULATIONS RELATING TO CONTAINMENT DESIGN

10 CFR Part 100, Reactor Sıte Crıterıa

The NRC staff has ın progress a study whıch would uncouple sıtıng requırements from specıfics of plant and contaınment desıgn In our report of June 13, 1990, we commented on thıs program and endorsed the general approach envısıoned by the staff

10 CFR Part 50, Appendıx J, Prımary Reactor Contaınment Leakage Testıng for Water-Cooled Power Reactors

If the allowable leakage rate for accıdent condıtıons ıs ıncreased, and ıf on-lıne monıtorıng capabılıty ıs provıded and used, the requırements of Appendıx J would have to be modıfied extensıvely Sıgnıficant sımplıficatıon of testıng requırements should be possıble

10 CFR 50 34(f) Addıtıonal TMI-Related Requırements

Addıtıonal requırements pertaınıng to contaınment desıgn were promulgated followıng the TMI-2 accıdent and are gıven ın 10 CFR 50 34(f) For example, a mınımum contaınment desıgn pressure of 45 psıg ıs specıfied ın one of these These requırements also apply to standard plant desıgns to be consıdered under 10 CFR Part 52 Some of these requırements would be superseded by the expanded GDC 50

10 CFR 50 44 Standards for Combustıble Gas Control System ın Lıght-Water-Cooled Power Reactors

Requırements ın thıs sectıon, ıntended for control of combustıble gas generated durıng severe accıdents, should be superseded by new GDC 50(c)

VII RECOMMENDED CONTENT OF THE NEW REGULATORY GUIDES FOR GENERAL DESIGN CRITERION 50

Implementatıon of our recommended GDC 50 wıll requıre development of new regulatory guıdes In what follows are some examples of what these guıdes should contaın In the final regulatory guıdes, to be developed by the staff, each should gıve acceptable values of the ımportant parameters as well as acceptable methods for theır calculatıon Realıstıc methods of calculatıon should be employed Our recommendatıons are keyed to the proposed GDC 50 [(a)-(h)]

50(a) Loss-of-Coolant Accıdent

Thıs regulatory guıde should address the current practıces for consıderıng LOCAS and ındıcate the followıng addıtıons or changes The best estımate methodology of 10 CFR 50 46 should be used Actıve coolıng systems should not be credıted ın calculatıng maxımum contaınment pressures durıng blowdown The effect of thermal stratıficatıon on thermal stresses ın steel lıners and penetratıons should be consıdered

50(b) Fuel–Coolant Interactıon In-Vessel FCI

Thıs regulatory guıde should treat PWRs and BWRs separately to account for dıfferences ın core degradatıon processes and dıfferıng amounts of zırcaloy relatıve to other materıals ın the core For PWRs wıth safety depressurızatıon systems or wıth low-pressure sequences of ımportance, ın-vessel FCI should be consıdered What constıtutes an acceptable mechanıstıc treatment to establısh the quantıtıes of molten core and ıts temperature should be delıneated Examples of acceptable methods for calculatıng the mechanıcal energy produced by FCI should be gıven Stıll further ıt should specıfy, for example, that mıssıle velocıty be calculated wıth consıderatıon gıven the vent path through the downcomer and possıble lower head faılure For present-day BWRs, ın-vessel FCI ıs not expected Future BWRs should be revıewed to be certaın thıs conclusıon ıs stıll valıd

Ex-Vessel FCI

For FCI outside the vessel, e g , in a water-filled cavity under the reactor, somewhat different assumptions would be appropriate Conditions at the time of vessel failure should be used to prescribe the amount and composition of the core material, and its temperature, that need to be considered for evaluation of ex-vessel FCI potential This regulatory guide should indicate what is acceptable as well as what is an acceptable method for its calculation

Containment designs that do not preclude water from being in the reactor cavity at the time of vessel failure must consider ex-vessel FCI This regulatory guide should indicate acceptable methods for calculation of the amount, composition and temperature of the molten core materials at the time of failure, and the type of vessel failure and mass flow rate of molten materials These calculated values should be used in calculating the mechanical energy produced by the FCI The mechanical energy calculation is to be based on the same method as described above

50(c) Hydrogen Combustion and Detonation

There will be different amounts of hydrogen generated by the different reactor types This regulatory guide should specify the amount of metals oxidized in-vessel and ex vessel as percentages of what is available, as well as give guidelines as to what constitutes an acceptable mechanistic method for calculating the rate and amount of hydrogen produced Hydrogen is produced following vessel failure during (1) interactions with water in the cavity, if it exists, and (2) subsequent corium–concrete interaction This regulatory guide should give guidance as to how much metal is oxidized in each of these two phases of the accident and give guidance to those who wish to calculate it themselves

Hydrogen in the containment atmosphere can lead to combustion, deflagration, or detonation All must be considered To deal with detonation, this regulatory guide should indicate what hydrogen control methods are acceptable and give both acceptable peak pressure and pressure pulse shape with guidance as to how they can be calculated This guide should also give examples of acceptable analysis methods for calculation of hydrogen distribution within the containment Pressure calculations should include the effect of hydrogen burns as well as carbon monoxide from corium–concrete interaction with account taken of the timing of the various gas generation processes The noncondensables from the corium–concrete interaction should also be considered in pressure calculations

50(d) Melt Attack on Containment Structure or Pressure Boundary

This regulatory guide should contain acceptable values of the molten core material composition, temperature, and rate at which it pours out of the vessel breach, as well as guidelines for an acceptable analysis Presence of water in the cavity under the reactor should be assumed if the plant is so configured If justified by a credible spreading analysis, uniform spreading may be assumed Otherwise, consequences of nonuniform melt depths should be considered Appropriate heat transfer calculations should be required to establish the thermal insult to the pressure boundary or essential structures

50(e) High Pressure Melt Ejection

This regulatory guide would apply only to PWRs and only if a depressurization system is not available It should give guidance on what constitutes acceptable analysis for calculation of thermal energy and corium composition shown to be credible at the time of failure of the reactor pressure vessel The regulatory guide should indicate that the amount, composition, flowrate, and temperature of the molten material be calculated by an acceptable method The containment atmosphere should be assumed to be saturated with water vapor Presence of water in the cavity under the reactor should be included in the analysis if the plant is so configured Allowable amounts of de-entrainment along the flow path should be specified or methods for their calculation should be given Oxidation of and heat transfer from the entrained debris should be based on mechanistic modeling

50(f) Corium–Concrete Interaction

This regulatory guide would be the same for all reactor types It should specify that a mechanistic evaluation of corium–concrete interaction be performed The results of an acceptable core melt and vessel failure analysis, defined in this guide, should be used to define the core melt characteristics as it arrives on the reactor cavity floor Water in the reactor cavity should be accounted for in calculations The basemat must be shown to be thick enough to provide an appropriate interdiction time before penetration With consideration given to timing, the contribution of combustibles and non-condensables to containment atmosphere pressure and temperature should be accounted for Selection of concrete types that reduce gas generation and the use of refractory materials should be encouraged Core debris control devices and filtered venting for long term pressure control should not be precluded by this guidance

50(g) Pressurization by Steam From Decay Heat

This regulatory guide should allow for credit to be taken for the decrease in decay heat with time, for heat transfer across the containment boundary, and for heat removal by operable containment equipment Restoration of emergency cooling should be credited after an appropriate time following the accident

50(h) Elevated Temperature

This regulatory guide should specify that a mechanistic calculation of the containment atmosphere thermal history be made with appropriate treatment of stratification including consideration of the following (1) hydrogen combustion, (2) high pressure melt ejection, (3) LOCAs, and (4) molten corium–concrete interaction A detailed heat transfer analysis should be required to ensure that seals, penetrations, equipment, and other items of safety significance are not damaged For containments with steel liners, thermal stresses induced by stratification should be considered

Final Rulemaking to Implement the Emergency Response Data System

Also at its May 1991 meeting, the ACRS discussed the NRC's proposed final rule that would amend 10 CFR Part 50 to establish requirements for the implementation of the Emergency Response Data System. The letter report reads, in part:[1]

> We previously commented on the proposed rule in our report of June 12, 1990 In that report, we did not support the proposed ERDS rule, although we acknowledged that it had some positive aspects
>
> During this meeting, we discussed Mr James M Taylor's July 24, 1990 response to the Committee's report in which he stated that the Commission, in approving SECY-80-433, had established the role and responsibility of the agency in nuclear plant accidents and that these have been articulated in NUREG-0728 and Manual Chapter 0502 He said also that the need for "timely, accurate and reasonably complete information about plant conditions, radiation releases and meteorological conditions at the site," as would be provided by ERDS, is fundamental in carrying out that role and that the ERDS rule would not change the NRC role or its responsibilities
>
> In addition, Mr Taylor stated that, based on his personal participation in actual responses to emergencies and exercises, "the risks of acting on inadequate or incorrect information are far greater than those associated with the modest amount of information that ERDS can provide"
>
> We were told by the staff and NUMARC that the voluntary program is not expected to result in industry-wide participation The present level of commitment represents about 55 percent of licensed power reactors, and is not expected to significantly increase without the rule
>
> As a result of our present review, we recommend that this rule be promulgated However, we continue to have a concern that ERDS might encourage inappropriate involvement of the NRC in the management of future serious accidents All operational aspects of accident management must be the responsibility of the licensee unless the Commission determines that formal intervention is necessary to protect the public health and safety
>
> We recommend that substantial experience be obtained with the operation of ERDS at a few plants before it is implemented industry-wide
>
> We have also observed that ERDS may not be available during loss-of-power events This suggests that emergency plan exercises should be carried out periodically without the availability of ERDS so that voice transmission of data can be practiced by participants

ACRS members W. Kerr and J. E. Wilkins, Jr., added the following additional comments:

> The Committee's report of June 12, 1990 did not support the proposed ERDS rule We still endorse that position and the justification therefor We recognize the staff's support and expressed need for the information that they believe will become available with the implementation of the ERDS However, our fear of inappropriate staff intervention in a serious and unanticipated severe accident continues to outweigh our evaluation of the benefits that might be provided by ERDS We therefore cannot endorse the rule

Additional comments were also provided by ACRS member H. W. Lewis:

> I continue to believe that the arguments made in our June 12, 1990 letter remain valid, and do not support this reversal on the part of the Committee Even the manual chapter on the division of responsibility between NRC and licensee in the event of a serious accident is ambiguous, opening the door to informal management on the part of both on-site and off-site NRC personnel A central principle of all emergency management is the need for an unambiguous chain of command, and a clear transfer of responsibility when management authority is transferred If this matter were clearly and unambiguously treated, I would see more merit in the proposed system ERDS is, after all, a direct descendant of the Nuclear Data Link, for which funds were long denied by the Congress, and which died for exactly these reasons

Proposed Final Rule Revising 10 CFR Part 55, "Operators' Licenses" to Include Fitness-for-Duty Requirements

At its May 8–11, 1991, meeting, the ACRS heard presentations concerning the staff's proposal to promulgate a final rule revising 10 CFR Part 55, "Operators Licenses," to include fitness-for-duty requirements and to modify Appendix C of 10 CFR Part 2, "General Statement of Policy and Procedures for NRC Enforcement Actions," to reflect enforcement sanctions. The letter report regarding these presentations reads, in part:[1]

> In our report of December 20, 1989, we concurred with the staff's plan to issue this proposed rule for public comment This proposed final rule includes the staff's evaluation of public comments
>
> This proposed rule, which the staff prepared in response to a Staff Requirements Memorandum dated March 22, 1989, would amend 10 CFR Part 55 so that the conditions and cutoff levels established pursuant to 10 CFR Part 26, "Fitness for Duty Programs," become applicable to licensed operators as a condition of their licenses The proposed rule will provide a basis for taking enforcement actions (as described in the proposed modifications to Appendix C of 10 CFR Part 2) against licensed operators who (1) use drugs or alcohol in a manner that would exceed the cutoff levels contained in the fitness-for-duty requirements of 10 CFR Part 26, (2) are determined by a facility medical review officer to be under the influence of any prescription or over-the-counter drug which could adversely affect his or her ability to safely and competently perform licensed duties, or (3) sell, use, or possess illegal drugs

We question the need for this rule. The fitness-for-duty requirements of 10 CFR Part 26 apply to all nuclear power plant personnel (including licensed operators), and the existing bases under 10 CFR Part 55 are available to the NRC for taking enforcement action against licensed operators for violation of fitness-for-duty requirements. Although there were nineteen Part 26 fitness-for-duty incidents involving licensed operators during 1990, the staff did not present any arguments that promulgation of this rule would have had an effect on this situation.

We are also concerned that promulgation of this rule will undercut industry's ongoing efforts to enhance the professionalism of all nuclear power plant personnel. The proposed rule appears to unnecessarily challenge the trustworthiness of licensed operators.

We recommend that this proposed rule not be issued. We believe that the industry has undertaken a substantial effort to deal with the difficult issue of fitness for duty and should be given the opportunity to demonstrate the effectiveness of its programs.

Proposed Final Revision to Appendix J to 10 CFR Part 50 and Related Final Regulatory Guide

The ACRS considered the proposed revision to Appendix J to 10 CFR Part 50, "Leakage Rate Testing of Containments of Light-Water-Cooled Nuclear Power Plants," and a related Regulatory Guide (Task No. MS 021-5), "Containment System Leakage Testing," at its May 8–11, 1991, meeting. The report reads, in part:[1]

We offer the following findings:

—Revision of Appendix J to 10 CFR Part 50 is desirable.

—The staff's proposal to make the revised version of Appendix J less prescriptive and to provide detailed guidance in a regulatory guide is appropriate.

—The implementation of the proposed revision to Appendix J clearly is a backfit.

—The staff has been unable to conclude that the proposed revision will provide a substantial increase in safety.

—The staff believes that the proposed revision will not increase costs to licensees; some licensees believe otherwise.

—There has been continuing constructive dialogue between the staff and industry representatives, chiefly relating to a Licensing Topical Report being prepared by the BWR Owners' Group. There are still some technical issues that would benefit from further dialogue between the staff and industry.

We understand from the staff that this dialogue will take place prior to issuance of this proposed revision.

—The proposed revision does not reflect new insights and knowledge about the role of containment, and containment leakage, in mitigating the consequences of severe accidents.

In view of these findings, we have no objection to the proposed revision to Appendix J to 10 CFR Part 50 or to the accompanying Regulatory Guide.

Documentation of Computer Codes

At its May 8–11, 1991, meeting, the ACRS again discussed the documentation requirements for computer codes. The letter report is cited here, in part:[1]

{We have been} asked to comment on a "Charter for Evaluation of RES Code Documentation." . . . In general, we believe the Charter is adequate. However, we recommend adding reference to NUREG-1230, Section 4.4.3, entitled "Code Documentation to Address Scaling and Code Applicability" so that the reviewers apply the lessons learned about documentation requirements from the TRACPF1/MOD1 uncertainty study.

We received a memorandum from Eric S. Beckjord, RES, to David A. Ward, ACRS, dated April 10, 1991, with an enclosure entitled "NRC/RES Software Documentation Guidance." Although this guidance is a beginning, it should be fleshed out by providing more explicit guidance concerning the contents of the "Code Manual" and the "Developmental Assessment" document. For example, the "Code Manual" should contain requirements for time-step and nodalization studies dealing with convergence and accuracy. The "Developmental Assessment" document should contain guidance for application of the codes to full-scale nuclear power plants with reference to the convergence and accuracy studies.

To summarize, we recommend the following:

1. The guidelines for code documentation supplied to us by RES should be fleshed out and cited by reference in all code development work statements. Programs to maintain existing codes should include a task to bring code documentation into compliance with the proposed guidelines.

2. A similar set of guidelines should be developed for use by NRR in its review of industry codes used for safety evaluations.

3. Our proposal to modify the Charter for Evaluation of RES Code Documentation review should be adopted.

Draft Final Rule on Nuclear Power Plant License Renewal

During its Apr. 11–13, 1991, meeting, the ACRS reviewed the draft of the final rule on nuclear power plant license renewal (10 CFR Part 54), which sets out the procedure and safety requirements that must be satisfied to renew an operating license of a commercial nuclear power plant beyond the initial 40-year license period. The pending draft specifies that applications for renewal could be made as much as 20 years before license expiration but not less than 3 years before a plant's authorization runs out. Although the first license is not due to expire until the year 2000, the industry views issuance of the rule as an important milestone to lay the groundwork for the filing of applications and initiation of the extensive reviews that will be required for license renewal.

The industry has a "lead plant" program to demonstrate the license renewal process. The program will establish the acceptable technical, environmental, and institutional justifications required to operate a nuclear power plant beyond its 40-year license term. Two nuclear power plants are preparing their applications for license renewal:[2]

—The Yankee Nuclear Power Station, a 185-MW pressurized-water reactor that started operating in 1960. Its license will expire in the year 2000.

—The Monticello Nuclear Generating Plant, a 545-MW boiling-water reactor that started operating in 1970. Its license expires in 2010.

The license renewal process is expected to take approximately two to three years once an application is submitted. The NRC has said that a plant would be allowed to continue to operate under the conditions of its current license while the application is under review. The NRC published a proposed rule for public comment in July 1990 and plans to issue a final rule at the end of June 1991. The Yankee Nuclear Power Station plans to submit its license renewal application in September 1991, and the Monticello Nuclear Generating Plant plans to submit its application in December 1991. The NRC expects to take two years to complete the review of a plant's application. The ACRS's comments on the draft final rule are given below.[3]

> The ACRS reported to you on the proposed license renewal rule in its report of April 11, 1990. Since that time, the proposed rule was published for public comment. The staff received 197 comments. It has assimilated information from these comments and information received in a number of interactions with industry and has prepared a draft final rule. . . . As stated in [that] report, we concur with the approach being taken by the staff in this rulemaking. However, there are two areas of disagreement between the staff and NUMARC that we would like to bring to your attention. The first might require a modification in the draft final rule. The second is related to implementation of the rule.
>
> The first matter is an issue on which we do not have a recommendation except that it should receive your consideration. The draft final rule requires that each applicant for license renewal develop a "compilation" of its Current Licensing Basis. Although it is not precisely clear what this means, it was agreed that it would, at a minimum, include a list of all licensing commitments agreed to by the applicant over the history of its plant. Industry representatives believe this is unnecessary.
>
> The second issue is how implementation of the rule will be limited in scope to concentrate resources for aging management where needed. The rule would require that each applicant develop a list of Systems, Structures, and Components Important to License Renewal [(SSCITLR)] and then implement an aging management program appropriate for items on that list. The staff's position is that the original SSCITLR list should include all those items in the plant that play a role in meeting any docketed commitment the licensee has made. This would include the original license; commitments related to new rules as they came into being; and commitments made in response to Safety Evaluation Reports, Information Notices, Bulletins, Generic Letters, and Orders.
>
> The industry representatives told us that such a definition of SSCITLR would result in a list that includes 85 to 90 percent of all equipment in the plant. They believe that application of a special aging program to all of these items would be unnecessary and onerous. The process of reducing the initial SSCITLR list to just those items to be covered by a special aging program is critical. Items important to implement other commitments would not thereby be ignored. They would be maintained through the new license period just as they are now.
>
> We believe that selection of those items to be subjected to a special aging program should be based on technical rather than legal argument. Our understanding is that a program of this nature can be developed with the rule as presently drafted. However, implementation will require careful crafting of the regulatory guide and the standard review plan. We would like the opportunity to review these documents before they are issued.

Staff Evaluation and Recommendations on Maintenance Rulemaking

At its Apr. 11–13, 1991, meeting, the ACRS discussed with the NRC staff their current evaluation and recommendations on maintenance rulemaking for nuclear power plants. The letter report reads, in part:[3]

> ACRS EVALUATION OF SECY-91-XXX
>
> —We are in agreement with the staff's assessment that the industry has made considerable improvement in the quality of nuclear power plant maintenance over the past several years. This is indicated by the results of maintenance team inspections, reinspections, and improving trends in performance indicators and SALP ratings.
>
> —We are impressed by ongoing industry initiatives and commitments to further improve nuclear power plant maintenance. These include the issuance of INPO 90-008, "Maintenance Programs in the Nuclear Power Industry," which is a compilation of INPO's maintenance perfor-

mance objectives and criteria The staff has reviewed INPO 90-008 and concluded that it is an acceptable industry maintenance program document delineating necessary program elements We agree that this document provides appropriate guidance to a utility manager on how to achieve the objectives required for a good maintenance program

—The draft Policy Statement, under "Maintenance Definition and Process," provides a compilation of "activities and supporting functions that should be considered in a maintenance program " This compilation comes from the staff's draft performance based regulatory guide and the Commission's current Policy Statement The listing uses language generally similar to but different from that of INPO 90-008 We recommend that this section of the Policy Statement either be deleted or revised to agree with INPO 90-008 in order to avoid confusion as to the Commission's views

—The draft Policy Statement, in the last paragraph under "Position," describes those structures, systems, and components that licensees should include in their maintenance programs We have two concerns with the language of the draft SECY document First, we believe that the scope envisioned for balance-of-plant SSCs is overly broad The staff told us that it has prepared revised wording to limit the scope for balance-of-plant SSCs to only those SSCs that could directly result in conditions adverse to safety This revised wording appears to be acceptable Our second concern is the absence of explicit language to require the inclusion in the scope of licensee's maintenance programs of those nonsafety-related SSCs that are important to the mitigation of severe accidents We recommend that the Policy Statement be revised to include these programs

—The staff told us that it plans to recommend that the maintenance escalation factor, which was made a part of the enforcement policy in the revised Policy Statement published on December 8, 1989, be rescinded [W]e disagreed with the original establishment of this escalation factor in our report of October 12, 1989 We agree with the staff that the maintenance escalation factor should be rescinded

—The staff plans to continue to monitor the effectiveness of licensee maintenance programs, as described under "Future Actions" in the draft Policy Statement This monitoring activity appears to be appropriate for the purpose

Proposed Policy Issues Identified in SECY-91-078 "Chapter 11 of the Electric Power Research Institute's Requirements Document and Additional Evolutionary Light Water Reactor Clarification Issues"

The ACRS discussed at its Apr. 11–13, 1991, meeting two Policy Issues identified in SECY-91-078 related to the certification of the Evolutionary Light Water Reactors. The report is cited, in part, as follows [3]

The staff's position regarding the first Policy Issue is that "an evolutionary ALWR design should include an alternate power source to the non-safety loads unless the design can demonstrate that the design margins in the evolutionary ALWR will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs " The staff's major concern is that the ALWR designs are departures from past practice and may result in an increased frequency of shutdowns that require cooling by natural circulation Presently licensed plants have electrical systems that provide an alternate power source to non-safety loads on shutdown However, the staff did not substantiate its concerns with respect to the proposed EPRI design requirements

EPRI claims that the ALWR is designed to safely accommodate shutdown with natural circulation and that the increased frequency of such events is small with this design The EPRI requirements for the ALWR electrical system design fully meet General Design Criterion 17, "Electric Power Systems," and the staff guidance contained in Regulatory Guide 1 32, Revision 2, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants " The ALWR electrical power system design is arranged to supply electric power to the plant's safety loads from the main generator, the plant switchyard, an independent transmission line, a gas turbine generator, and the diesel generators The design uses a generator circuit breaker between the main generator and the step-up transformer and has an improved full turbine load rejection capability EPRI claims high reliability of electric power to the unit auxiliary transformers and has provided data to support its claim that the benefits derived from adding an alternate power source to the non-safety loads are small and not cost effective We concur with the EPRI position

The staff's position regarding the second Policy Issue is based on a misunderstanding of the text of the EPRI requirements As a result, the staff proposes an additional requirement that "at least one offsite circuit to each redundant safety division should be supplied directly from one of the offsite power sources with no intervening non-safety buses, in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus " The staff's concern is that routing offsite power to the safety buses through non-safety buses may subject safety equipment to undesirable disturbances on the non-safety buses Therefore, the staff's position would require the capability to supply safety buses directly from offsite power The staff did not substantiate its concern However, the EPRI requirements for ALWR electrical power system design already provide one alternate circuit to each of the redundant safety divisions directly from offsite power This meets the staff's position EPRI agreed to clarify the text to document this requirement EPRI's position is that the direct circuit from offsite to each of the redundant safety divisions should be the backup power supply and the normal supply should be from the plant's auxiliary electric system We concur with EPRI's position, but do not believe *that this should become a regulatory requirement*

Proposed Resolution of Generic Safety Issue 130, "Essential Service Water System Failures at Multi-Unit Sites"

ACRS reviewed the proposed resolution of Generic Safety Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," and issued a report which reads, in part:[3]

> We do not agree with the staff's conclusion that issuance of the proposed generic letter has been justified on a cost–benefit basis. A number of assumptions used in the analysis do not appear to provide a fair and balanced comparison of potential costs and benefits. It appears to us that there would be a wide variation in the conclusions if the analysis were done for each individual plant.
>
> We believe that the emergency service water systems of these seven plants should be analyzed as a part of their Individual Plant Examinations. Vulnerabilities should be corrected where necessary. The staff should consider making the analysis it has performed for this proposed resolution available to these licensees for use in performing their IPEs.
>
> In the interim, we believe that the staff can assure itself through its inspection program that the licensees of these plants are applying appropriate risk management to the operation and surveillance of their emergency service water systems.

FEDERAL COURT RULES TO GIVE "PUBLIC CITIZEN" ACCESS TO NUCLEAR INDUSTRY RECORDS

The U.S. Court of Appeals for the District of Columbia sided with the organization "Public Citizen" in its seven-year lawsuit against the NRC to gain access to nuclear industry safety records.[4] The appeal involved a claim under the Freedom of Information Act (FOIA) regarding certain reports dealing with nuclear power plant safety. The disputed reports were prepared by the Institute for Nuclear Power Operations (INPO), a utility consortium. Since 1982, INPO has shared these reports with NRC pursuant to an agreement providing for the free exchange of nuclear information. The Critical Mass Energy Project, the energy policy arm of Public Citizen, has been trying to secure copies of the reports from the Commission under the FOIA.

In 1984, NRC refused to release the INPO reports to Critical Mass, citing FOIA's exemption for confidential commercial information. Critical Mass then brought suit at two lower court levels challenging the agency's action, which resulted in the case coming to the Appeals Court. The Court of Appeals found "no basis to conclude that disclosure of the relevant reports would be likely to result in any significant impairment of either the effectiveness or the efficiency of the NRC by virtue of anticipated antagonism in the relationship between INPO and the Commission." Therefore the court reversed the judgment of the District Court on that point. The Court of Appeals also remanded the case to the District Court for further proceedings to resolve several other questions of fact "consistent with this opinion."

"The decision is a major step toward the public finally gaining access to key information on the safety records of the nation's nuclear power plants," asserted K. Bossong, director of Public Citizen's Critical Mass Energy Project. "The long fight waged by the nuclear industry and the NRC to keep this information secret strongly suggests that the records will confirm that the plants are far more dangerous than they would like to admit."

NRC said that it was studying the court's decision and planned to continue defending its position in accordance with the FOIA exemption for confidential commercial information.

LICENSING REFORM LEGISLATION DEBATED IN SENATE SUBCOMMITTEE

Two proposed pieces of legislation are in the Senate that would, among other goals, change the way the NRC would license nuclear plants in the future. S. 341, introduced by Sen. J. B. Johnston (D-La.) and Sen. M. Wallop (R-Wyo.), and S. 570, introduced by the Administration, both contain provisions that permit only one public hearing at the construction phase of a plant. The licensing provisions of these bills and the issue of whether NRC has statutory authority to make its own rule were the subject of debate at a Senate subcommittee hearing on nuclear regulation in mid-May 1991 (Ref. 5).

Subcommittee Chairman Sen. R. Graham (D-Fla.) opened by noting that one of the biggest problems faced by the nuclear industry is the protracted and expensive plant licensing process. Graham stated that licensing should not delay the construction of new nuclear plants and pledged to consider legislation to eliminate the second public hearing now required prior to operation. Sen. A. K. Simpson (R-Wyo.) declared that the "parade of witnesses" protesting new licensing regulations was only "pretending to protect the public." At the heart of their protests, suggested Simpson, is an attempt to thwart the development of nuclear energy. These remarks were apparently directed chiefly toward E. R. Glitzenstein, a lawyer representing the Union of Concerned Scientists

(UCS) at the hearing. UCS has vehemently protested the passage of either bill, arguing that they "would completely undermine public participation in the licensing process" and would also "hamper the NRC's own ability to safeguard the public's health and safety."

The NRC itself previously submitted legislation requesting Congress to allow changes to the licensing process. Specifically, NRC asked that the Atomic Energy Act be amended to facilitate issuance of a combined construction permit and operating license, early site permits, and approval of standardized designs. Congress did not act on that request, and it was then, said J. M. Taylor, executive director of operations at NRC, that the Commission decided that the proposals could be implemented under its own authority. At the suggestion of various congressional committees, NRC then instituted a rulemaking that resulted in the promulgation of 10 CFR Part 52.

As described by NRC, Part 52 establishes a combined construction permit–operating license proceeding that is meant to resolve most licensing issues before construction. After construction, NRC said, the public could still contest issuance of authorization to operate by filing a petition that the Commission must, by law, respond to. This new rule stirred up much opposition, and the Nuclear Information and Resource Service (NIRS) asked for judicial review of the rule in a Court of Appeals. The petition was granted, and the court upheld most of the rule but reversed the part limiting the scope of the post-construction hearing. This amounted to a small victory for NIRS, even though the court granted NRC the right to a rehearing, which was to take place later in 1991.

Taylor noted that "the Commission does not object to being given additional authority to streamline the licensing process." He said that NRC "fully supports the goals" of both S. 341 and S. 570 but does have questions–problems with some parts of the licensing titles. Specifically, the question of exactly when adjudicatory procedures should be used during the licensing process concerns the Commission.

Under S. 341 as marked up, and under S. 570 as proposed, post-license construction, safety, and emergency planning issues would be treated as petitions to modify the license. No adjudicatory hearing would be required. These two facts are what rankle groups like UCS. Glitzenstein argues that S. 341 as marked up and amended by Wallop "abolishes the affected public's fundamental right to a hearing on whether the nuclear plant constructed in their neighborhood is sufficiently safe for operation to begin." S. 570 is even worse, said UCS counsel. It includes a provision that "even if the Commission does decide to hold a hearing on a major violation of the combined license—and even if it finds that this violation involves serious health risks—the Commission is not 'permitted' to provide for 'discovery and cross-examination of witnesses.' "

On May 23, 1991, the Senate Energy Committee approved S. 341 by a 17 to 3 vote.[6]

NRC PROBLEM FACILITIES LIST UPDATED[7]

In mid-June 1991 NRC chairman K. M. Carr was briefed by NRC staff and regional administrators on the status of reactors and fuel facilities on the Commission's problem plant list.

Two of the Tennessee Valley Authority's (TVA) Browns Ferry units, 1 and 3, remain on the Category 3 list, which comprises shutdown plants requiring NRC authorization before a restart and close NRC monitoring. Category 2, composed of plants authorized to operate but which NRC will monitor more closely than the average plant, lists TVA's Browns Ferry 3; Calvert Cliffs 1 and 2, owned by Baltimore Gas and Electric; and Zion 1 and 2 licensed to Commonwealth Edison Company.

Category 1 represents the plants that have been removed from the list of problem facilities; this status has been achieved by Niagara Mohawk's Nine Mile Point units 1 and 2.

Three plants, in addition to Nine Mile Point 1 and 2, were recognized for achieving a high level of safety performance in multiple areas. These are Union Electric Company's Callaway plant, Northern States Power Company's Prairie Island plant, and GPU Nuclear Corporation's TMI Unit 1.

In addition to nuclear reactors, attention was also paid to four nuclear materials enrichment plants that were classified as priority facilities, meaning subject to enhanced NRC attention. They are GE's enrichment plant in Wilmington, N.C., Nuclear Fuel Services' facility in Erwin, Tenn., Safety Light Corporation's Bloomsburg, Pa., plant, and Sequoyah Fuels Corporation's facility in Gore, Okla.

General Electric's Wilmington facility earned its place on this list by an accident on May 28, 1991, which was reported by plant officials who identified the presence of approximately 150 kg of uranium suspended in liquid solution in a tank. A critical reaction could have occurred if enough uranium in the solution had precipitated to the bottom of the tank. Implementing emergency procedures, plant workers and NRC experts averted the potential criticality safety problem. As a result of the accident, plant-wide procedures were undergoing NRC review, and the possibility of stationing a resident NRC inspector at the plant was under consideration.

At Sequoyah Fuels, liquid uranium hexafluoride and uranium tetrafluoride were discovered to have leaked underground at the site. As much as 9 600 kg (21 000 lb) of liquids was estimated to have accumulated in the earth under the plant. According to NRC, efforts are currently under way to study the possibility of groundwater contamination, the company has committed to new health physics studies, and activities on new regulatory actions are also under way.

NRC ISSUES REGULATIONS TO ESTABLISH A DISCRETIONARY APPEAL REVIEW SYSTEM [8]

In an action that became effective on July 25, 1991, the NRC amended its procedural rules to establish a system in which, instead of mandatory Atomic Safety and Licensing Appeals Board (ASLAB), the Commission may, if it so desires, review appeals against most decisions by its Atomic Safety and Licensing Boards (ASLB) and Administrative Law Judges (ALJ). Only appeals of certain decisions regarding high-level radioactive waste will require mandatory review.

The new system replaces mandatory reviews formerly provided by the ASLAB, which was abolished in October 1990. Under the new system, parties may petition for direct Commission review of ASLB and ALJ decisions. If the Commission chooses not to review a decision, it becomes a final agency action (still subject, however, to a review by a Federal Court of Appeals). The filing of a petition for Commission review is a necessary step before a party may seek judicial review. The Commission also proposed to establish a new Office of Commission Appellate Adjudication to assist it in exercising its adjudicatory responsibilities.

CONSTRUCTION APPROVAL GIVEN FOR WORLD'S FIRST ADVANCED NUCLEAR POWER PLANT [9]

Construction of the world's first advanced-design nuclear power plants, a two-unit facility, was approved to begin in late Summer 1991 in Japan, according to GE Nuclear Energy, headquartered in San Jose, Calif. GE will provide the reactors, nuclear fuel, and turbine generators for the two units.

Japan's Ministry of International Trade and Industry formally announced the granting of the "Establishment Permit" to Tokyo Electric Power Company for installation of the first advanced boiling-water reactor (ABWR) plants at its Kashiwazaki-Kariwa nuclear power station northwest of Tokyo.

The plants are characterized by simplified design and improved reliability and safety as well as reduced construction, fuel, and operating costs. The first unit is scheduled for start of commercial operation in July 1996, with the second unit to follow one year later.

The granting of the "Establishment Permit" successfully concludes a safety review of the ABWR design by Japan's Atomic Energy Commission and Nuclear Safety Commission. Included were extensive safety analyses and public hearings that confirmed the safety of the ABWR.

The ABWR is a 1356-MW(e) plant; it was selected as the next generation boiling-water reactor (BWR) in Japan where nuclear power currently accounts for 28% of electricity generation. Japan's goal is to generate more than 40% of its electricity from nuclear energy by the end of the next decade.

In the United States the GE ABWR was expected to become the first certified standard nuclear plant design in 1992, under the U.S. plant standardization program.

The two ABWRs will be the sixth and seventh units at the Kashiwazaki-Kariwa site, where three BWRs of an earlier design are now in operation and where two more are under construction.

NRC DENIES PETITION FOR SHUTDOWN OF YANKEE ROWE NUCLEAR PLANT

In late June 1991 the NRC denied a petition by the Union of Concerned Scientists (UCS) and the New England Coalition on Nuclear Pollution (NECNP) asking for an immediate shutdown of the Yankee Rowe Nuclear Power Station in Massachusetts.[10] The UCS and NECNP claimed that the plant was in violation of NRC requirements for reactor pressure-vessel integrity and that an NRC staff safety assessment of the plant performed in August 1990 was flawed.[11]

In a letter denying the petition, Dr. T. Murley, NRC's Director of Nuclear Reactor Regulation, stated that "staff has evaluated the Yankee Rowe reactor vessel issues carefully and has concluded that the vessel condition continues to provide adequate protection of the public health and safety." When the safety assessment was performed in August of last year, NRC determined that the plant, which is the oldest in the country, could operate safely until April 1992. At that time a "comprehensive vessel examination" and review of data from a research program sponsored by the licensee was to be conducted by the NRC.

Though denying the petition, NRC staff did acknowledge that Yankee Rowe has not met regulatory requirements in several different areas.[12] The petitioners were most concerned with the integrity of the reactor pressure vessel, which has been embrittled by fast-neutron irradiation from the reactor within it. Pressurized thermal shock testing led NRC staff to conclude that "the PTS screening criterion may have been exceeded." In his letter, Murley stated "that belief is based on conservatively considering the uncertainties associated with weld chemistry, irradiation temperature, grain size effects and flaw distribution as noted" in the previous staff assessment. Although conceding that there could be a problem with higher than acceptable reference temperatures for the upper and lower plates and the circumferential weld, Murley pointed out that "10 CFR 50.61 does not require shutdown if the PTS screening criterion is exceeded." On an individual case-by-case basis, NRC may allow operation of a reactor at reference temperatures that exceed PTS screening criterion, he said.

The UCS and NENCP were wholly unsatisfied with the NRC response to their petition. In a press statement released the same day that NRC's denial was made public, UCS announced that it would take the matter to court. "We are disappointed but not surprised," said E. Quinn of UCS, who was optimistic about the petition's chances in court, because, he said, "the regulations are numerical here." He noted that there is a history of appeals courts granting petitions that NRC turned down.

Adding fuel to the fire on the side of the petitioners were Sen. E. M. Kennedy (D-Mass.) and Sen. P. J. Leahy (D-Vt.), and Reps. E. J. Markey (D-Mass.) and B. Sanders (I-Vt.). In their letter to NRC Chairman Carr they requested the "expeditious review of information which raises serious safety concerns at the Yankee Rowe nuclear power plant." The NRC ruling was only preliminary, in response to the request for an immediate shutdown of the plant. Unofficially, both sides agreed that the pending, more-detailed response would differ little in substance.

REFERENCES

1. NRC Press Release 91-52, May 23, 1991.
2. *At Energy Clearing House*, 37(24): 5 (June 14, 1991).
3. NRC Press Release 91-46, Apr. 30, 1991.
4. *At Energy Clearing House*, 37(18): 2 (May 3, 1991).
5. *At Energy Clearing House*, 37(20): 4 (May 17, 1991).
6. *At Energy Clearing House*, 37(22): 3 (May 31, 1991).
7. *At Energy Clearing House*, 37(25): 1 (June 21, 1991).
8. NRC Press Release 91-65, June 25, 1991.
9. *At Energy Clearing House*, 37(21): 2 (May 24, 1991).
10. *At Energy Clearing House*, 37(26): 1 (June 28, 1991).
11. NRC Press Release 91-66, June 25, 1991.
12. *At Energy Clearing House*, 37(26): 1 (June 28, 1991).

Reports, Standards, and Safety Guides

By D. S. Queener[a]

This article contains four lists of various documents relevant to nuclear safety as compiled by the editor. These lists are: (1) reactor operations-related reports of U.S. origin, (2) other books and reports, (3) regulatory guides, and (4) nuclear standards. Each list contains the documents in its category which were published (or became available) during the three-month period (April, May, and June 1991) covered by this issue of *Nuclear Safety*. The availability and cost of the documents are noted in most instances.

OPERATIONS REPORTS

This category is listed separately because of the increasing interest in the safety implications of information derivable from both normal and off-normal operating experience with licensed power reactors. The reports fall into several categories shown, with information about the availability of the reports given where possible. The NRC reports are available from the Nuclear Regulatory Commission (NRC) Public Document Room, 1717 H Street, Washington, DC 20555, for inspection, or photocopies can be obtained from the NRC Public Document Room at a fee of $0.05 per page, minimum charge $2.00.

NRC Office of Nuclear Reactor Regulation

The NRC Office of Nuclear Reactor Regulation (NRR) issues reports regarding abnormal occurrences at licensed reactors. These reports, previously published by the NRC Office of Inspection and Enforcement (IE), fall into two categories of urgency: (1) NRC Bulletins, which require remedial actions and/or responses from affected licensees, and (2) NRC Information Notices, which are for general information and do not require any response.

NRC Information Notices

NRC IN 86-21, Supplement 2 *Recognition of American Society of Mechanical Engineers Accreditation Program for N Stamp Holders,* April 16, 1991, 3 pages plus 3 pages of attachments.

NRC IN 88-63, Supplement 2 *High Radiation Hazards from Irradiated Incore Detectors and Cables,* June 25, 1991, 4 pages plus 2 pages of attachments.

NRC IN 89-01, Supplement 2 *Failure of Westinghouse Steam Generator Tube Mechanical Plugs,* June 28, 1991, 6 pages plus one-page attachment.

NRC IN 91-25 *Commercial-Grade Structural Framing Components Supplied as Nuclear Safety-Related Equipment,* April 1, 1991.

NRC IN 91-26 *Potential Nonconservative Errors in the Working Format Hansen–Roach Cross-Section Set Provided with the Keno and Scale Codes,* April 2, 1991.

NRC IN 91-27 *Incorrect Rotation of Positive Displacement Pump,* April 10, 1991, 2 pages plus one-page attachment.

NRC IN 91-28 *Cracking in Feedwater System Piping,* April 15, 1991.

NRC IN 91-29 *Deficiencies Identified During Electrical Distribution System Functional Inspections,* April 15, 1991, 4 pages plus one-page attachment.

NRC IN 91-30 *Inadequate Calibration of Thermoluminescent Dosimeters Utilized to Monitor Extremity Dose at Uranium Processing and Fabrication Facilities,* April 23, 1991, 3 pages plus 3 pages of attachments.

NRC IN 91-31 *Nonconforming Magnaflux Magnetic Particle (14AM) Bath,* May 9, 1991, 2 pages plus 9 pages of attachments.

NRC IN 91-32 *Possible Flaws in Certain Piping Systems Fabricated by Associated Piping and Engineering,* May 15, 1991, 3 pages plus one-page attachment.

NRC IN 91-33 *Reactor Safety Information for States During Exercises and Emergencies,* May 31, 1991, 2 pages plus one-page attachment.

NRC IN 91-34 *Potential Problems in Identifying Causes of Emergency Diesel Generator Malfunctions,* June 3, 1991, 3 pages plus one-page attachment.

NRC IN 91-35 *Labeling Requirements for Transporting Multi-Hazard Radioactive Materials,* June 7, 1991, 2 pages plus one-page attachment.

NRC IN 91-36 *Nuclear Plant Staff Working Hours,* June 10, 1991, 3 pages plus one-page attachment.

NRC IN 91-37 *Compressed Gas Cylinder Missile Hazards,* June 10, 1991, 3 pages plus one-page attachment.

NRC IN 91-38 *Thermal Stratification in Feedwater System Piping,* June 13, 1991, 3 pages plus 2 pages of attachments.

NRC IN 91-39 *Compliance with 10 CFR Part 21, Reporting of Defects and Noncompliance,* June 17, 1991, 3 pages plus 8 pages of attachments.

NRC IN 91-40 *Contamination of Nonradioactive System and Resulting Possibility for Unmonitored, Uncontrolled*

[a]Oak Ridge National Laboratory.

Release to the Environment June 19, 1991, 3 pages plus 2 pages of attachments

NRC IN 91 41 *Potential Problems with the Use of Freeze Seals* June 27, 1991, 3 pages plus one-page attachment

NRC IN 91 42 *Plant Outage Events Involving Poor Coordination Between Operations and Maintenance Personnel During Valve Testing and Manipulations* June 27, 1991, 3 pages plus one page attachment

Other Operations Reports

These are other reports issued by various organizations in the United States dealing with power reactor operations activities As of May 8, 1985, the NRC no longer sells its publications as a sales agent for the U S Government Printing Office (GPO) However, most of the NUREG series documents can be ordered from the Superintendent of Documents, U S Government Printing Office, P O Box 37082, Washington, DC 20013 A number of these reports can also be ordered from the NRC Public Document Room Specify the report number when ordering Telephone orders can be made by calling (202) 275-2060

Many other reports prepared by U S Government laboratories and contractor organizations are available from the National Technical Information Service (NTIS), U S Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161, and/or DOE Office of Scientific and Technical Information, P O Box 62, Oak Ridge, TN 37831 General Accounting Office (GAO) reports can be obtained at no charge for single copies from U S GAO, Document Handling and Information Services Facility, P O Box 6015, Gaithersburg, MD 20760 Reports available through one or more of these organizations are designated with the appropriate information (i e , GAO, GPO, NTIS, and OSTI) in parentheses at the end of the listing, followed by the price, when available

NUREG-0090, Vol 13, No 4 *Report to Congress on Abnormal Occurrences October–December 1990* March 1991, 17 pages (GPO)

NUREG-0090 Vol 14, No 1 *Report to Congress on Abnormal Occurrences January–March 1991* June 1991, 13 pages (GPO)

NUREG-1421 *Regulatory Analysis for the Resolution of Generic Issue 130 Essential Service Water System Failures at Multi Unit Sites* V Leung et al , June 1991, 25 pages (GPO)

NUREG/CR 2000, Vol 10, No 3 *Licensee Event Report (LER) Compilation for Month of March 1991* April 1991, 90 pages (GPO)

NUREG/CR-2000, Vol 10 No 4 *Licensee Event Report (LER) Compilation for Month of April 1991* May 1991, 85 pages (GPO)

NUREG/CR 2000, Vol 10, No 5 *Licensee Event Report (LER) Compilation for Month of May 1991* June 1991, 87 pages (GPO)

NUREG/CR 2907 *Radioactive Materials Released from Nuclear Power Plants Annual Report 1988* J Tichler et al , Brookhaven National Lab , NY, May 1991, 315 pages (GPO)

NUREG/CR 5456 *Analysis of Flow Stratification in the Surge Line of the Comanche Peak Reactor* J G Sun et al , Argonne National Lab , IL April 1991, 50 pages (GPO)

NUREG/CR-5706 *Potential Safety-Related Pump Loss An Assessment of Industry Data NRC Bulletin 88-04* D A Casada, Oak Ridge National Lab , TN, June 1991, 43 pages (GPO)

OTHER BOOKS AND REPORTS

During April, May, and June 1991, the following selected safety-related books and reports became available Included are publications which were not received under foreign exchange agreements and which do not deal directly with U S power-reactor experiences The documents in this list obtainable from U S Government distribution organizations are designated by the appropriate code in parentheses, as described for the list of "Other Operations Reports" immediately preceding

DOE- and NRC-Related Items

NUREG-1125, Volume 12 *A Compilation of Reports of The Advisory Committee on Reactor Safeguards 1990 Annual* April 1991, 141 pages (GPO)

NUREG-1374 *Technical Findings Related to Generic Issue 79 An Evaluation of PWR Reactor Vessel Thermal Stress During Natural Convection Cooldown* J D Page, May 1991, 145 pages (GPO)

NUREG-1394 *Emergency Response Data System (ERDS) Implementation* J Jolicoeur, June 1991, 45 pages (GPO)

NUREG-1407 *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities Final Report* J T Chen et al , June 1991, 75 pages (GPO)

NUREG-1435, Vol 3 *Status of Safety Issues at Licensed Power Plants Generic Safety Issues* June 1991 (GPO)

NUREG/CR 4893 *Technical Findings Report for Generic Issue 135 Steam Generator and Steam Line Overfill Issues* Scientech Inc , ID, May 1991, 102 pages (GPO)

NUREG/CR-5467 *Risk Based Inspection Guide for Crystal River Unit 3 Nuclear Power Plant* B W Smith et al , Pacific Northwest Lab , WA, June 1991, 60 pages (GPO)

NUREG/CR-5543 *A Systematic Process for Developing and Assessing Accident Management Plans* D J Hanson et al , Idaho National Engineering Lab ID, April 1991, 85 pages (GPO)

NUREG/CR-5550 *Passive Nondestructive Assay of Nuclear Materials,* D. Reilly et al., Los Alamos National Lab., NM, March 1991, 700 pages (GPO).

NUREG/CR-5611 *Issues and Approaches for Using Equipment Reliability Alert Levels,* E. V. Lofgren and S. H. Gregory, Brookhaven National Lab., NY, June 1991 (GPO).

NUREG/CR-5682 *Specific Topics in Severe Accident Management,* J. F. Meyer et al., Scientech Inc., MD, May 1991 (GPO).

NUREG/CR-5692 *Generic Risk Insights for General Electric Boiling Water Reactors,* R. Travis et al., Brookhaven National Lab., NY, May 1991, 76 pages (GPO).

Other Items

EPRI NP-7045M *Hydrogen Combustion Experiments in 1/4-Scale Model of Mark II Nuclear Reactor Containment,* F. Tamanini et al., Electric Power Research Institute (EPRI), CA, January 1991 [available from EPRI Research Reports Center (RRC), Box 50490, Palo Alto, CA 94303].

NSAC-138 *Validation of RETRAN-03 from FIST Turbine Trip Data and Boil-Off Data,* C. E. Peterson and J. L. Westacott, EPRI, CA, April 1991, 51 pages (EPRI-RRC).

NSAC-166 *Losses of Off-Site Power at U S Nuclear Power Plants Through 1990,* H. Wyckoff, EPRI, CA, April 1991, 115 pages (EPRI-RRC).

World Nuclear Performance Through April 1991, Vol. 6, No. 6, McGraw-Hill Nuclear Publications, June 1991, 55 pages (McGraw-Hill Inc., 1221 Avenue of the Americas, New York, NY 10020).

IAEA/WCRT/SRA/1 *Water Cooled Reactor Technology Safety Research Abstracts, No 1,* International Atomic Energy Agency (IAEA), Vienna, December 1990, 1093 pages (UNIPUB, 4611-F Assembly Drive, Lanham, MD 20706-4391).

Disposal of Radioactive Waste, Heterogeneity of Groundwater Flow and Site Evaluation, Proceedings of an NEA Workshop, Paris, October 22–24, 1990, OECD Nuclear Energy Agency, Paris, 1991, 335 pages (OECD Publications and Information Center, 2001 L Street NW, Suite 700, Washington, DC 20036-4910).

Nuclear Energy Data 1991, OECD, 1991, 44 pages (OECD).

The Interface in Nuclear Safety and Public Health, Proceedings of an NEA Seminar, Paris, September 12–13, 1990, OECD, 1991, 250 pages (OECD).

REGULATORY GUIDES

To expedite the role and function of the NRC, its Office of Nuclear Regulatory Research prepares and maintains a file of Regulatory Guides that define much of the basis for the licensing of nuclear facilities. These Regulatory Guides are divided into 10 divisions as shown in Table 1.

Table 1 Regulatory Guides

Division 1,	Power Reactor Guides
Division 2,	Research and Test Reactor Guides
Division 3,	Fuels and Materials Facilities Guides
Division 4,	Environmental and Siting Guides
Division 5,	Materials and Plant Protection Guides
Division 6,	Product Guides
Division 7,	Transportation Guides
Division 8,	Occupational Health Guides
Division 9,	Antitrust and Financial Review Guides
Division 10,	General Guides

Single copies of draft guides may be obtained from NRC Distribution Section, Division of Information Support Services, Washington, DC 20555. Draft guides are issued free (for comment), and licensees receive both draft and final copies free; others can purchase single copies of Active Guides by contacting the U.S. Government Printing Office, Superintendent of Documents, P.O. Box 37082, Washington, DC 20013. Costs vary according to length of the guide. Of course, draft and active copies will be available from the NRC Public Document Room, 1717 H Street, NW, Washington, D.C., for inspection and copying for a fee.

Revisions in these rates will be announced as appropriate. Subscription requests should be sent to the National Technical Information Service, Subscription Department, Springfield, VA 22161. Any questions or comments about the sale of regulatory guides should be directed to Chief, Document Management Branch, Division of Technical Information and Document Control, Nuclear Regulatory Commission, Washington, DC 20555.

Actions pertaining to specific guides (such as issuance of new guides, issuance for comment, or withdrawal), which occurred during the April, May, and June 1991 reporting period, are listed below.

Division 1 Power Reactor Guides

Draft Regulatory Guide DG-1008 *Reactor Coolant Pump Seals,* April 1991.

Division 5 Materials and Plant Protection Guides

Regulatory Guide 5.66 *Access Authorization Program for Nuclear Power Plants,* June 1991.

Division 7 Transportation Guides

Regulatory Guide 7.11 (previously Drafts MS 144-4 and DG-7001) *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels With a Maximum Wall Thickness of 4 Inches (0 1 m),* June 1991.

Regulatory Guide 7.12 (previously Drafts MS 501-4 and DG-7002) *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels With a Wall Thickness Greater than 4 Inches (0 1 m) But Not Exceeding 12 Inches (0 3 m)*, June 1991

NUCLEAR STANDARDS

Standards pertaining to nuclear materials and facilities are prepared by many technical societies and organizations in the United States, including the Department of Energy (DOE) (NE Standards). When standards prepared by a technical society are submitted to the American National Standards Institute (ANSI) for consideration as an American National Standard, they are assigned ANSI standard numbers, although they may also contain the identification of the originating organization and be sold by that organization as well as by ANSI. We have undertaken to list here the most significant nuclear standards actions taken by organizations during April, May, and June 1991. Actions listed include issuance for comments, approval by the ANSI Board of Standards Review (ANSI-BSR), and publication of the approved standard. Persons interested in obtaining copies of the standards should write to the issuing organizations.

American National Standards Institute

ANSI does not prepare standards; it is devoted to approving and disseminating standards prepared by technical organizations. However, it does publish standards, and such standards can be ordered from ANSI, Attention: Sales Department, 1430 Broadway, New York, NY 10018. Frequently, ANSI is an alternate source for standards also available from the preparing organization.

BSR N665-1985 (withdrawal of ANSI N665-1985, for comment) *Nuclear Fuel Facilities—Facilities or Fabricating Fuel for Light Water Reactors (LWR)—Fire Protection*, $15.00.

BSR N304-1986 (withdrawal of ANSI N304-1986, for comment) *Nuclear Fuel Facilities—Facilities for Reprocessing Fuel—Fire Protection*, $23 00.

BSR/ASME N278 1-1975 (Reaffirmation and Redesignation of ANSI N278.1-1975, R1985, for comment) *Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard*, $8 00.

ANSI N317-1980 (R1991, Reaffirmation, approved by ANSI/BSR) *Performance Criteria for Instrumentation Used to Inplant Plutonium Monitoring*

ANSI N42.18-1980 (R1991, Reaffirmation, approved by ANSI/BSR) *Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents*

ANSI N46.1-1980 (Withdrawal, approved by ANSI/BSR) *Guidance for Defining Safety-Related Features of Nuclear Fuel Cycle Facilities*

ANSI N305-1975 (R1981, withdrawal, approved by ANSI/BSR) *Design Objectives for Highly Radioactive Solid Material Handling and Storage Facilities in a Reprocessing Plant*

American Nuclear Society

Standards prepared by ANS can be obtained from ANS, Attention: Marilyn D. Weber, 555 North Kensington Avenue, LaGrange Park, IL 60525.

ANSI/ANS 51.10-1991 (Revision of ANSI/ANS 51.10-1979, approved by ANSI/BSR) *Auxiliary Feedwater System for Pressurized Water Reactors*

American Society of Mechanical Engineers

Standards prepared by ASME can be obtained from ASME, Attention: D. Palumbo, 345 East 47th Street, New York, NY 10017.

BSR/ASME NQA-2B (Revision of ANSI/ASME NQA-2-1989, for comment) *Quality Assurance Requirements for Nuclear Facility Applications*, $20 00.

ANSI/ASME NQA-1B-1991 (Addenda to ANSI/ASME NQA-1-1989, Published) *Quality Assurance Program Requirements for Nuclear Facilities*

BSR/ASME QME-1 (New Standard, approved by ANSI/BSR) *Section QR, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants*, $9 00

Proposed Rule Changes as of June 30, 1991[a,b]

(Changes Since the Previous Issue of *Nuclear Safety* Are Indicated by Shaded Areas)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 0 10 CFR 1 10 CFR 2			7-29-91	Procedures for direct Commission review of decisions of presiding offices	Final rule in 56:124 (29403)
10 CFR 1	12-12-88	1-30-89		Policy statement on exemptions from regulatory control	Advanced notice of proposed policy statement in 53:238 (49886)
10 CFR 2	9-26-89	11-27-89	2-26-91; 3-28-91	Procedures applicable to proceedings for the issuance of licenses for the receipt of high-level radioactive waste at a geologic repository	Published for comment in 54:185 (39387); correction in 54:190 (40780); final rule in 56:37 (7787); corrections in 56: 66 (14151)
10 CFR 2	4-3-90	6-18-90		Revisions to procedures to issue orders	Published for comment in 55:64 (12370)
10 CFR 2	7-5-90	9-4-90		Revisions to procedures to issue orders: challenges to orders that are made immediately effective	Published for comment in 55:129 (27645)
10 CFR 2 10 CFR 50 10 CFR 54	7-17-90	10-15-90		Nuclear power plant license renewal	Published for comment in 55:137 (29043); request for extension of comment period denied in 55:166 (34939)
10 CFR 2	10-24-90	12-10-90		Options and procedures for direct Commission review of licensing board decisions	Published for comment in 55:206 (42947)
10 CFR 2 10 CFR 40 10 CFR 70 10 CFR 74	12-17-90	3-4-91		Material control and accounting requirements for uranium enrichment facilities producing special nuclear material of low strategic significance	Published for comment in 55:242 (51726)
10 CFR 2 10 CFR 19 10 CFR 20 10 CFR 30 10 CFR 31 10 CFR 32 10 CFR 34 10 CFR 35 10 CFR 39 10 CFR 40 10 CFR 50 10 CFR 61 10 CFR 70			6-20-91	Standards for protection against radiation	Final rule in 56: 98 (23360); corrections in 56:101 (23956)

Proposed Rule Changes as of June 30, 1991 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 4	3-8-89	5-8-89		Enforcement of nondiscrimination on the basis of handicap in federally assisted programs: notice of proposed rulemaking	Published for comment in 54:44 (9966); corrections in 54:51 (11224)
10 CFR 13	9-25-90	11-24-90		Program Fraud Civil Remedies Act; implementation	Published for comment in 55:186 (39158); corrections in 55:194 (40997)
10 CFR 16	9-26-90	10-26-90		Salary offset procedure for collecting debts owed by federal employees to the federal government	Published for comment in 55:187 (39285)
10 CFR 19 10 CFR 20 10 CFR 21 10 CFR 30 10 CFR 36 10 CFR 40 10 CFR 51 10 CFR 70 10 CFR 170	12-4-90	3-4-91		Licenses and radiation safety requirements for large irradiators	Published for comment in 55:233 (50008)
10 CFR 20 10 CFR 30 10 CFR 40 10 CFR 70	5-14-90	7-30-90		Notification of incidents	Published for comment in 55:93 (19890)
10 CFR 20 10 CFR 21 10 CFR 73			4-26-91	Change in commercial telephone numbers for Region V	Final rule in 56:81 (19253)
10 CFR 20			5-10-91	Petitions requested below regulatory concern exemptions	Deferral of action published in 56: 91 (21631)
10 CFR 26	8-31-90	10-30-90		Fitness-for-Duty Programs: nuclear power plant personnel	Published for comment in 55:170 (35648)
10 CFR 30 10 CFR 40 10 CFR 50 10 CFR 60 10 CFR 61 10 CFR 70 10 CFR 72 10 CFR 110 10 CFR 150	4-3-90	6-18-90		Willful misconduct by unlicensed persons	Published for comment in 55:64 (12374); corrections in 55:70 (13542)
10 CFR 35	1-16-90	4-12-90		Basic quality assurance program, records and reports of misadministration or events relating to the medical use of byproduct material	Published for comment in 55:10 (1439); corrections in 55:25 (4049)

(Table continues on the next page.)

Proposed Rule Changes as of June 30, 1991 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 50	3-6-89	7-5-89		Acceptance of products purchased for use in nuclear power plant structures, systems, and components	Published for comment in 54:42 (9229)
10 CFR 50	10-13-89	12-1-89		Nuclear power plant license renewal	Published for comment in 54:197 (41980)
10 CFR 50	10-9-90	12-24-90		Emergency response data system	Published for comment in 55:195 (41095)
10 CFR 50	1-31-91	4-16-91		Codes and standards for nuclear power plants	Published for comment in 56:21 (3796)
10 CFR 50			6-14-91	Fracture toughness requirements for protection against pressurized thermal shock events	Final rule in 56:94 (22300)
10 CFR 51	7-23-90	10-22-90		License renewal for nuclear power plants; scope of environmental effects	Advanced notice of proposed rulemaking published for comment in 55:141 (29964)
10 CFR 55	4-17-90	7-2-90		Operator's licenses	Published for comment in 55:74 (14288)
10 CFR 70 10 CFR 72 10 CFR 73 10 CFR 75	8-15-89	9-29-89		Minor amendments to the physical protection requirements	Published for comment in 54:156 (33570)
10 CFR 71	6-8-88	10-6-88; 12-6-88; 3-6-89; ~6-15-89;[c] 2-9-90		Transportation regulations; compatibility with the International Atomic Energy Agency (IAEA)	Published for comment in 53:110 (21550); corrections published in 53:120 (23484); comment period extended in 53:190 (38297); 2nd extension of comment period in 53:245 (51281); 3rd extension of comment period in 54:63 (13528); comment period end published in 54:237 (51033)
10 CFR 73			4-25-91[d]	Access Authorization Program for nuclear power plants	Final rule in 56:80 (18997); corrections in 56:103 (24239)
10 CFR 110	2-7-90	3-9-90		Import and export of radioactive wastes	Advance notice of proposed rulemaking for comment in 55:26 (4181); corrections in 55:57 (10786)
10 CFR 110			5-31-91	Return of Topaz reactor to Soviet Union	Final rule in 56:105 (24682)

Proposed Rule Changes as of June 30, 1991 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
48 CFR 20	10 2 89	12-1-89		Acquisition regulation (NRCAR)	Published for comment in 54 189 (40420)

[a]NRC petitions for rule making are not included here, but quarterly listings of such petitions can be obtained by writing to Division of Rules and Records, Office of Administration, U S Nuclear Regulatory Commission, Washington, DC 20555 Quarterly listings of the status of proposed rules are also available from the same address

[b]Proposed rules for which the comment period expired more than 2 years prior to the start of the period currently covered without any subsequent action are dropped from this table Effective rules are removed from this listing in the issue after their effective date is announced

[c]The expiration date is given as "60 days after the date when the DOT proposed rule is published in the *Federal Register*"

[d]Except for the information collection requirements in 73 56(a) (1), (2), and (3), 73 56(b) (1) and (2), 73 56(c), 73 56(d), 73 56(e), 73 56(f) (1) and (2), and 73 56(h) (1) These will become effective upon OMB approval, NRC will publish the effective dates in the *Federal Register*

The Authors

Report on the International Symposium on the Use of Probabilistic Safety Assessment for Operational Safety—PSA '91

Sabyasachi Chakraborty is Nuclear Safety Research Manager and Assistant to the Director of the Swiss Federal Nuclear Safety Inspectorate. He is also a consultant to the European Space Agency on reviewing ongoing development of Procedure Guides on Hazard Analysis and Hazard and Risk Reduction, Risk Assessment, and Risk Analysis Data Collection Method. He received the B.S degree in physics from the Calcutta University in India and holds a Diploma in Physics degree from the University of Kiel in Germany. While at the Swiss Federal Nuclear Safety Inspectorate, he led the development of Safety Guides on Emergency Planning for the International Atomic Energy Agency (IAEA), and he has lectured on probabilistic safety assessment and accident management under the sponsorship of IAEA and the Organization for Economic Cooperation and Development, Nuclear Energy Agency He has served as the Technical Program Committee Chairman of the International Conferences on PSA'87, PSA'91, and recovery operations in the event of a nuclear accident or radiological emergency. He has published a considerable number of scientific papers in the area of severe accident and risk analysis associated with nuclear power plants. He is the coauthor of a book on worldwide consequences of the Chernobyl accident and the editor of *Risk Analysis as a Decision Tool* and *Integral View on Risks—Technical, Ethical and Social Aspects* (all in German, published in Germany by TUeV Rheinland Publishing Company) Current address: Swiss Federal Nuclear Safety Inspectorate, CH–5232 Villigen-HSK, Switzerland.

Mohsen Khatib-Rahbar is president of Energy Research, Inc. His research interests include nuclear reactor dynamics, thermal hydraulics, severe accident phenomenology, safety, and probabilistic risk assessment. He has served as a consultant to several organizations, including the U.S. Nuclear Regulatory Commission, U S. Department of Energy, Pacific Northwest Laboratories, Los Alamos National Laboratory, the Swiss Federal Nuclear Safety Inspectorate, and the International Atomic Energy Agency. Before founding Energy Research, Inc., he was a member of the scientific staff of Brookhaven National Laboratory He has also served as a visiting scientist at GRS in Germany (1982) and the U S. Nuclear Regulatory Commission Office of Research (1988–1989). He received a B.S. degree in chemical engineering from the University of Minnesota in 1974 and a Ph.D degree in nuclear science and engineering from Cornell University in 1978. Current address: Energy Research, Inc., 6290 Montrose Rd., Suite 200, Rockville, MD 20852.

Good Relationships Are Pivotal in Nuclear Data Bases

A. Sharif Heger is an assistant professor in the Nuclear Engineering Department of the University of New Mexico He received his Ph D. degree in nuclear engineering in 1989, his M.S. degree in nuclear engineering in 1983, and his B.S. degree in mechanical engineering in 1976, all from the University of Texas He has an active research program in probabilistic risk assessment and belief networks. For his doctoral research he developed a prototype of an adaptive data-base interface (KNOWBOT) The original version of the system was based on associative information retrieval. A new version of the system based on belief networks is under development for adaptive journal article search and retrieval based on the faculty research interest profile. Another system is also under development for Los Alamos National Laboratory for finding patterns of concern in its operating experience data base He is a member of American Nuclear Society. Current address· The University of Texas, Mechanical Engineering Department, Austin, Tex.

Billy V. Koen has been a professor of nuclear engineering at the University of Texas at Austin since 1968 He received the Sc.D. and S.M. degrees in nuclear engineering from the Massachusetts Institute of Technology, a Diplome d'Ingenieur from L'Institut National des Sciences et Techniques Nucleaires at Saclay, France, the B.S. degree in chemical engineering and the B A. degree in chemistry from the University of Texas at Austin He was a consultant in nuclear reactor reliability at Saclay, France, in 1971–1972 and 1976–1977 and has taught at Ecole Centrale Koen is a member of the American Nuclear Society and the Association des Ingenieurs en Genie Atomique. Current address· The University of New Mexico, Chemical/Nuclear Engineering Dept., Albuquerque, New Mex.

Technical Note: The Interagency Nuclear Safety Review Panel's Evaluation of the Ulysses Space Mission

Lieutenant Colonel Joseph A. Sholtis, Jr. received the M.S. degree in nuclear engineering from the University of New Mexico in 1977. He served as Department of Defense (DOD) Co-Coordinator for the Interagency Nuclear Safety Review Panel evaluation of the Ulysses space mission. Current address: Directorate of Nuclear Security, Air Force Safety Agency, Kirtland AFB, New Mexico 87117-5000.

Darrell A. Huff received the B.S. degree in applied nuclear physics from Brigham Young University in 1976. He was the Department of Energy (DOE) Coordinator for the Interagency Nuclear Safety Review Panel evaluation of the Ulysses space mission. Current address: Office of Environment, Safety and Health, U.S. DOE, Washington, DC 20545.

Leven B. Gray received the B.S. degree in physics from the University of Missouri in 1956. He was the National Aeronautics and Space Administration (NASA) Coordinator for the Interagency Nuclear Safety Review Panel evaluation of the Ulysses space mission. Current address: Office of Safety and Mission Quality, NASA Headquarters, Washington, DC 20546.

Norman P. Klug received the M.S. degree in nuclear engineering from the University of Michigan in 1959 and the M.A. degree in general administration from the University of Maryland in 1984. He was the DOE Co-Coordinator for the Interagency Nuclear Safety Review Panel evaluation of the Ulysses space mission. Current address: Office of Light Water Reactor Safety and Technology, U.S. DOE, Washington, DC 20545.

Colonel Robert O. Winchester received the M.S. degree in nuclear engineering from the Air Force Institute of Technology in Dayton, Ohio, in 1969. He was the DOD Co-Coordinator for the Interagency Nuclear Safety Review Panel evaluation of the Ulysses space mission. Current address: Directorate of Nuclear Society, Air Force Safety Agency, Kirtland AFB, New Mexico 87117-5000.

The Severe Accident Analysis Program for the Savannah River Nuclear Production Reactors

M. L. Hyder is a Senior Advisory Chemist with the Westinghouse Savannah River Company at the Savannah River Site near Aiken, S.C. He joined the research staff at the Savannah River Laboratory (SRL) in 1962 after receiving the Ph.D. degree in chemistry from the University of California at Berkeley. His research interests during his career at SRL have included radiation chemistry, analytical chemistry, chemical separations processes, and, most recently, the chemical phenomena involved in severe reactor accidents. He is currently the technical leader of the Severe Accident Analysis Program. Current address: Westinghouse Savannah River Company, SRL, Aiken, SC 29808.

A Framework for Selecting Suitable Control Technologies for Nuclear Power Plant Systems

Roger A. Kisner is Manager of the Control Engineering Group with the Instrumentation and Controls (I&C) Division at the Oak Ridge National Laboratory (ORNL). The Control Engineering Group is responsible for research, development, design, and implementation of control technologies on ORNL projects as well as projects for other organizations, such as the Electric Power Research Institute, the Nuclear Regulatory Commission, Babcock and Wilcox, General Electric, and Duke Power. His background includes the development, design, and installation of a wide range of control, protection, and monitoring systems for industrial processes and commercial systems. This background also includes research in human–machine interactions. He is a technical editor of the Control and Instrumentation Section of *Nuclear Safety*. Other technical topics of interest include fault-tolerant control system architectures; automated plant startup strategies; learning, adaptive, and self-tuning control techniques; control system human–machine interface; and effects of electromagnetic interference and other environmental effects on control system electronics. He is a registered Professional Engineer and holds B.S. and M.S. degrees in nuclear science and engineering from Virginia Polytechnic Institute and State University. Current address: ORNL, Instrumentation and Controls Division, Oak Ridge, TN 37830-6010.

Containments for Gas-Cooled Power Reactors: History and Status

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Fort St. Vrain, the Gas-Cooled Fast Breeder Reactor, the large standardized Gas-Cooled Reactor (GASSAR), and the Modular High Temperature Gas-Cooled Reactor (MHTGR). Earlier he was a research associate in the Department of Aerospace and Mechanical Sciences, Princeton University, and also worked for private industry. He holds a B.S. degree in chemical engineering from Cornell University, an M.S. degree in nuclear engineering from Massachusetts Institute of Technology, and a Ph.D. degree in nuclear engineering from the University of Maryland. Current address: Director, HTGR Division, NE45, U.S. Department of Energy, Washington, DC 20545.

Indoor Radon: A Natural Risk

Naomi H. Harley is a Research Professor in the Department of Environmental Medicine at the New York University Medical Center. She received the B.S.E.E. degree from the Cooper Union and the M.E. and Ph.D. degrees from New York University. She worked for a number of years at the Health and Safety Laboratory of the Atomic Energy Commission (AEC) on radon, natural radioactivity, and weapons test fallout. She presently teaches several radiation courses at the University while doing research on radon and other natural radionuclides. She is also very active in preparing reports for the National Council on Radiation Protection and Measurements. Current address: P.O. Box M-268, Hoboken, NJ 07030.

John H. Harley is now a consultant on radioactivity after retiring as director of the Environmental Measurements Laboratory (EML) in 1980. He received the Sc.B. degree in chemistry from Brown University and the M.S. and Ph.D. degrees from Rensselaer Polytechnic Institute. During World War II he was involved in analytical chemistry work for the Manhattan Project at Union Carbide Research Laboratories in Niagara Falls, New York. He joined the Atomic Energy Commission (AEC) Health and Safety Laboratory (HASL) as director of the Analytical Division and became Laboratory director in 1960. Under the Department of Energy, formerly AEC, HASL later became EML. Current address: P.O. Box M-268, Hoboken, NJ 07030.

On the Importance of the Atmospheric Parameters in the Fission Products Distribution of a Severe Reactor Accident

Mahmut Celal Barla is on the Aeronautics and Astronautics faculty at the Technical University of Istanbul and at the Maritime High School. He has been an Associate Professor since 1986. He worked as a research assistant at the Toulouse III University (1982–1983) and at Arlon—Belgium (1985–1986). He is a specialist of cloud physics and has worked at the University since 1973. Current address: Technical University of Istanbul, 80626 Maslak, Istanbul, Turkey.

Ahmet Bayülken is on the teaching staff of the Institute of Nuclear Energy at the Technical University of Istanbul. He received the M.Sc. degree in mechanical engineering from the same university in 1970. He did his Ph.D. thesis at the Sorbonne University in Paris in 1976 and has worked at the Institute since 1970. He was deputy director of the Cekmece Nuclear Research Center from 1983 to 1987 and became a Professor in 1989. He is also one of the supervisors of the TRIGA Mark-II Research Reactor at the Institute. He is main interests are in the economics of nuclear power plants, radiological protection, and safety. Current address: Technical University of Istanbul, 80626 Maslak, Istanbul, Turkey.

Book Review: *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V*

Chester R. Richmond: Current address: Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Effects of Component Aging on the Westinghouse Control Rod Drive System

Ken Sullivan is a research engineer at Brookhaven National Laboratory (BNL), operated for the U.S. Department of Energy (DOE) by Associated Universities Incorporated. He received his Bachelor of Electrical-Computer Technology from the New York Institute of Technology. Before joining the Plant Systems Evaluation Group within BNL's Department of Nuclear Energy, he worked for 8 years in the Instrumentation and Controls Group at BNL's High Flux Beam Reactor, where his responsibilities included evaluating and testing process and control instrumentation systems. In addition to providing technical assistance to both the U.S. Nuclear Regulatory Commission (NRC) and DOE, his current activities include system performance evaluations in support of the Nuclear Plant Aging Research Program sponsored by NRC. Current address: BNL, Building 130, Upton, NY 11973.

Bill Gunther is a research engineer at Brookhaven National Laboratory (BNL). After obtaining B.S. and

M S degrees in electrical engineering from Northeastern University, he worked for 15 years at an electric utility and was involved with the construction, startup, and operation of commercial nuclear power plants, where he maintained a senior reactor operator's license He is currently responsible for a variety of research and technical assistance activities sponsored by the U S Nuclear Regulatory Commission and the U S Department of Energy He is a registered professional engineer in the state of New York Current address BNL, Building 130, Upton, NY 11973

CORRECTION

In our Issue 32(2) Prof Dr Jan Madey (current address Institute of Informatics, Warsaw University, ul Banacha 2, 00-913 Warsaw, POLAND) was inadvertently referred to as "she ' Prof Madey informs us that he "is a man, and always has been!" We regret the error

Reviewers of *Nuclear Safety,* Volume 32

The technical quality of a journal depends not only on the competence and efforts of its authors and editorial staff but also, to a major extent, on the dedication of its corps of peer reviewers. We wish to acknowledge gratefully the many technical experts whose voluntary and unrewarded reviews of proposed *Nuclear Safety* articles have been indispensable in the selection of articles and in the revision of articles to prepare them for publication.

We list below all the names of those who reviewed articles for publication in Vol. 32, whether the articles were used or not. Since it is our policy not to reveal the reviewers' identities to the authors, all reviewers are listed in alphabetical order together with their affiliations.

This list does not include, though we are most grateful to them also, the names of the DOE and NRC staff members who review all *Nuclear Safety* articles to assure that the policies and positions of their agencies are not misstated or distorted.

Alexander, C., Battelle Columbus Laboratories, Columbus, Ohio

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Corradini, M. L., University of Wisconsin, Madison, Wis.

Dilworth, G. F. (2), Technology Resources International, Knoxville, Tenn.
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Hanna, S., Sigma Research Corporation, Westford, Mass.
Haugh, J. J., Electric Power Research Institute, Palo Alto, Calif.
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Lupton, L , Chalk River Laboratories, Ontario, Canada

Malinauskas, A P , Oak Ridge National Laboratory, Oak Ridge, Tenn
Martin, W R , Oak Ridge National Laboratory, Oak Ridge, Tenn
Mears, L D , Gas-Cooled Reactor Associates, San Diego, Calif
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Miller, C W , Department of Nuclear Safety, Springfield, Ill
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Muller, K F , Westinghouse Electric Corporation, Pittsburgh, Pa
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Wilson, J N , U S Nuclear Regulatory Commission, Washington, D C
Witherspoon, J P , Consultant, Oak Ridge, Tenn

Zmola, P C , Consultant, Bethesda, Md

NEW OECD "INTERNATIONAL INFORMATION SYSTEM ON OCCUPATIONAL EXPOSURE (ISOE)"

The following information has been received from the Radiation Protection and Waste Management Division of the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-Operation and Development (OECD):

The OECD Nuclear Energy Agency (NEA), based in Paris, launched in 1988 the idea of creating an international system for information exchange in the field of occupational exposure. The System would allow the integration of experience gained in nuclear power plant operation in the various OECD Countries for the purpose of facilitating operational dose management and optimisation of radiation protection. This idea, which has been approved by the NEA Committee on Radiation Protection and Public Health, led to the official launching of the ISOE System on the occasion of the first meeting of its Steering Group, on 18 November 1991 in Paris.

The primary objective of the ISOE is to make available to its participants a computerized tool to accelerate the dissemination of operational experience in radiation protection of workers and facilitate exchange of information between utilities on specific dosimetric problems and new maintenance operations with which they are confronted.

All types of power reactors existing in the OECD Countries will be included in the ISOE (PWR, BWR, GCR and Candu). The information in the System will concern the following three data bases, defined as NEA1, NEA2 and NEA3:

a) various performance indicators of special interest to radiation protection, such as collective dose, individual dose distribution (NEA1) and job related doses, information which will be updated annually;

b) information about methods and techniques used for effective dose control, such as information about material used in specific components, chemical specifications for process water and procedures for training and work planning (NEA2); this type of information, which varies little from one year to another, will be updated in case of change;

c) brief descriptions of specific operations carried out and radiation protection problems faced with (NEA3); this level, which will also contain information about contact persons at the nuclear power plants, will be continuously updated all around the year and represents a main interest of the system.

The technical operations of the ISOE will be carried out by three Technical Regional Centres, one in North America, one in Europe and one in Japan.

From a practical point of view it is worth mentioning that each participant will receive the ISOE data management and analysis software, developed for microcomputers (type PC 386 or PC 2). The Technical Regional Centres will then periodically provide participants with the updated data base on floppy discs.

From the beginning of its operation, the ISOE will have participation from Belgium, France, Italy, the Netherlands, Canada, Germany, Spain, Sweden, Finland, Japan, and the United Kingdom.

The United States is expected to join in the near future. The decision to participate has been delayed due to the large number of utilities which have to be consulted. The International Atomic Energy Agency (IAEA) and the Commission of the European Communities (CEC) are participating in the System. With the CEC, that runs a system of data collection on job related doses, a special cooperation agreement has been prepared allowing complete information exchange between the parts that are in common between the two systems.

The first meeting of the Steering Group elected Mr. Ph. Rollin, EdF, France, chairman of ISOE and Mr. A. Khan, Ontario Hydro, Canada, vice-chairman. The Bureau of the Steering Group was completed with Mr. P. O'Donnell, from the Spanish safety authorities, CSN.

For further information, please contact a Member of the Bureau: M. Ph. Rollin, France, Telephone (1) 40 42 50 40; Mr. A. Khan, Canada, Telephone (416) 683 7516; Sr. P. O'Donnell, Spain, Telephone (1) 346 0561; or contact the NEA Secretariat: M. C. Viktorsson, France, Telephone (1) 45 24 96 26.

HARVARD SCHOOL OF PUBLIC HEALTH ANNOUNCES SHORT COURSES

Boston, Mass.

The School of Public Health of Harvard University announces the following short courses to be offered in the summer of 1992

Radiation Protection Instrumentation, May 11–15, 1992. This course covers calibration and evaluation of beta and gamma survey instruments, evaluation of "hot particles," assessment of dry active waste, measurements of mixed radiation fields, whole body counting, and sampling and analysis of airborne radionuclides

In-Place Filter Testing Workshop, June 8–12, 1992. This 5-day lecture and practicum program provides laboratory and nuclear air cleaning professionals with an in-depth understanding of air filtration theory, aerosol technology, air flow measurements, and in-place testing of particulate (HEPA) filters and gas adsorption units The course includes laboratory sessions and review of NRC regulations Emphasis is placed on practice in the use of instruments

Environmental Radiation Surveillance, June 8–12, 1992. This offering will focus on internal and external radiation standards, biological effects and epidemiological studies, environmental transport models, measurement, data interpretation, quality assurance, and the design of surveillance programs

Planning for Nuclear Emergencies, June 15–19, 1992. This program provides detailed coverage of scenario development, accident source terms and dose estimates, standards and guides for emergency response, training and notification systems, protective action guides (PAGs), the roles of state and federal agencies, public health needs, and working with public information agencies and the media NUREG-0654 will receive particular attention

For further information on these and other programs, please contact Mary F McPeak, Office of Continuing Education, Harvard School of Public Health, 677 Huntington Ave , Boston, MA 02115 Telephone 617-432-1171 Fax 617-432-1969

EIGHTH POWER PLANT DYNAMICS, CONTROL AND TESTING SYMPOSIUM

Knoxville, Tenn., May 27–29, 1992

This symposium, sponsored by the Department of Nuclear Engineering of the University of Tennessee College of Engineering, will be held at the Hyatt Regency Hotel in Knoxville, Tenn It will cover the topics of power plant simulation, new trends in modeling methods, plant testing for model validation and parameter estimation, training simulators, expert systems and artificial intelligence control applications, plant monitoring and diagnostics, neural networks applications, experimental methods in liquid–metal reactors, advances in data acquisition, analysis, and signal validation, instrument fault detection and estimation, and methods for predictive maintenance There will be focused topical sessions on international developments in nuclear power plant instrumentation and control and on future directions in power plant instrumentation and control

For complete information, contact Dr B R Upadhyaya or Dr E M Katz, Department of Nuclear Engineering, The University of Tennessee, Knoxville, TN 37996-2300 Telephone (615) 974-5048 Fax (615) 974-0668

SECOND TRAINING COURSE ON OFF-SITE EMERGENCY PLANNING AND RESPONSE FOR NUCLEAR ACCIDENTS

Mol, Belgium, June 29–July 3, 1992

The CEC is promoting education and training activities in radiation protection (ERPET: European Radiation Protection Education and Training) to maintain and extend Community expertise in radiation protection.

The accidents at Three Mile Island and Chernobyl triggered a worldwide effort reviewing the hardware and software aspects of the off-site response to nuclear accidents. This training course is intended to provide a comprehensive understanding of all aspects of today's emergency planning and response, i.e., principles of intervention, planning and organization, and decision making with respect to off-site intervention in the case of an accidental release of radioactive material to the environment.

The course is intended for those involved in contingency planning, e.g., civil protection and environmental protection officers, persons responsible for the management of radiation protection or emergency planning at nuclear facilities. The course will also provide general background to junior health physicists and other people whose prime responsibilities are not directly in this field but are interested in a general review of its present trends.

The list of topics to be covered includes: experience of past accidents, potential accident scenarios, transfer to the environment, exposure pathways, health aspects, accident consequences, remedial actions, intervention criteria, emergency planning reference accidents, real-time assessments, environmental monitoring, decision aiding techniques, information of the public, organization of emergency plans, and emergency response exercises.

Case studies, exercises, and discussions will familiarize the participants with the problems to implement principles. Lectures will be given by staff of the SCK/CEN, CEC and other European Institutes.

For further information, contact Mrs. E. Van Gelder, Administrative Secretary, Studiecentrum voor Kernenergie, B-2400 Mol, Belgium. Telephone: (32) 14 33 21 11, Ext. 5244. Fax: (32) 14 32 10 56.

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SUBSCRIPTIONS

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FUTURE ARTICLES

GENERAL SAFETY

The Twentieth NRC Water Reactor Safety Information Meeting, *M. D. Muhlheim*

ACCIDENT ANALYSIS

Analysis and Modelling of Fission Product Release from Heated Uranium–Aluminum Plate-Type Reactor Fuels, *R. P. Taleyarkhan*

INSTRUMENTATION AND CONTROLS

Application of a Surveillance and Diagnostics Methodology Using Neutron Noise From a Pressurized Water Reactor, *R. T. Wood, L. F. Miller, and R. P. Perez*

Upgrading of the K-Reactor Supplementary Safety System at the Savannah River Site, *L. R. Canas, N. T. Hightower III, I. K. Palik, and L. A. Wooten*

DESIGN FEATURES

Safety Aspects of the Requirements for the Passive Advanced Light Water Reactor, *T. U. Marston, W. H. Layman, and G. Bockhold, Jr.*

Westinghouse Advanced Passive 600 Plant, *B. A. McIntyre and R. K. Beck*

ENVIRONMENTAL EFFECTS

The MATS Experiments—Mesoscale Atmospheric Transport Studies at the Savannah River Site, *A. H. Weber, S. Berman, R. J. Kurzeja, and R. P. Addis*

OPERATING EXPERIENCE

Summary of the Fuel Performance Annual Report for 1989, *F. M. Berting*

Experience With After-Shutdown Decay Heat Removal—BWRs and PWRs, *J. J. Haugh, F. J. Mollerus, and H. R. Booth*