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NUCLEAR SAFETY

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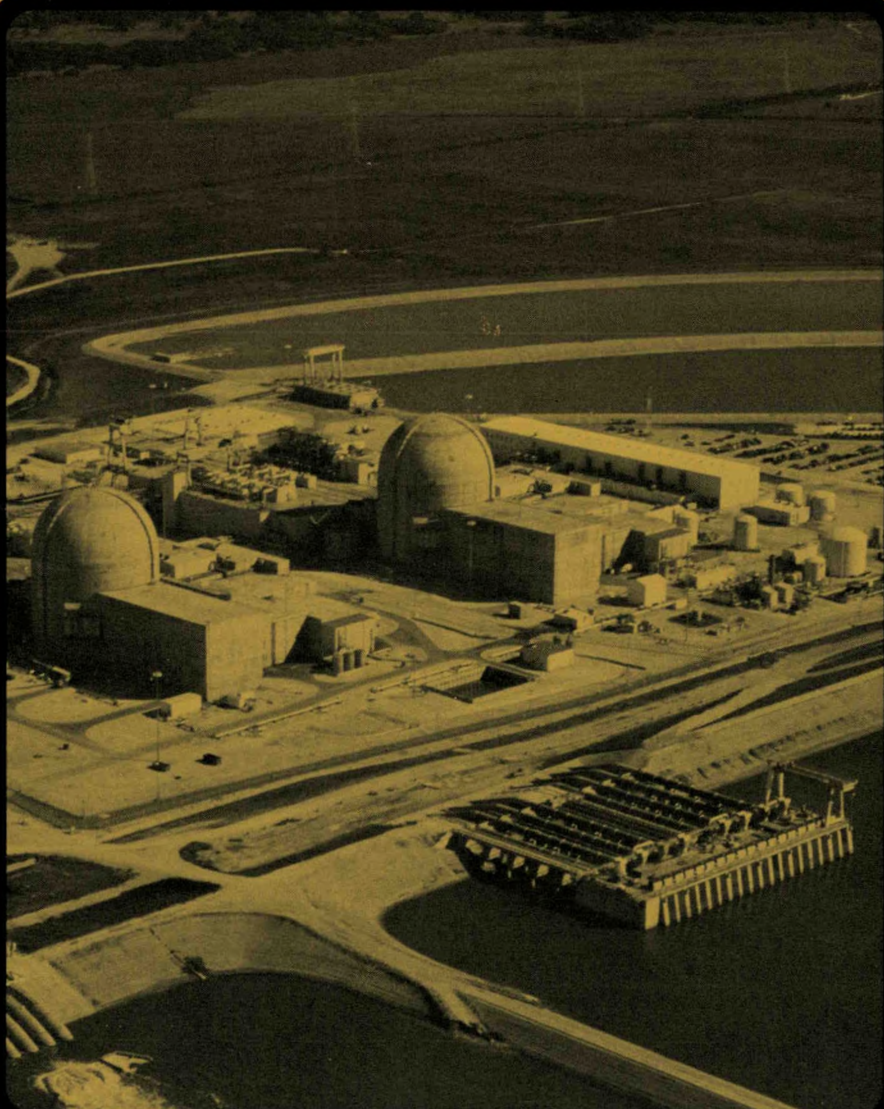

TECHNICAL PROGRESS REVIEW

JUL • SEP 1989

NUREG/CR-4674
ORNL/NOAC-232
Vol. 7

Precursors to Potential Severe Core Damage Accidents: 1987 A Status Report

Main Report and Appendix A

Prepared by J. W. Minarick, J. D. Harris, J. W. Cletcher, P. N. Austin, A. A. Blake

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

The Nuclear Operations Analysis Center (NOAC) of the Oak Ridge National Laboratory has prepared this latest member of a series of reports, whose coverage goes back to 1969, as part of its ongoing Accident Sequence Precursor Program. This program reviews licensee event reports (LERs) of operational events to identify and categorize precursors to potential severe core-damage accidents. Such precursors are infrequent initiating events or equipment failures that, had additional subsequent failures also occurred, could have resulted in a plant condition with inadequate core cooling. In other words, they are events that proceeded part-way on an identified path of multiple failures that could potentially lead to a severe core-damage accident but did not do so because the later failures did not occur. This report is available from the National Technical Information Service, Springfield, Va. 22161 or the NRC/GPO Sales Program, Superintendent of Documents, Government Printing Office, Washington, DC 20402.

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 Joel R. Buchanan, Director, NOAC, P.O. Box 2009, Oak Ridge National Laboratory, Oak Ridge, TN 37831-8065. Telephone (615) 574-0393 (FTS: 624-0393).

Cover: As reported in this issue of *Nuclear Safety*, the Houston Lighting and Power Company received a full-power license for South Texas 2, a 1250-MW(e) pressurized-water reactor located on the Colorado River in Matagorda County, Tex. Both sister units are shown in this aerial photograph of the station.

A quarterly Technical Progress Review
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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 not usually exceed 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), Environment, Safety and Health, Office of Safety Appraisals, 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.

Editorial Staff

General Safety Considerations

Edited by J. R. Buchanan

Safety of Framatome Advanced Nuclear Steam Supply Systems Designs

By J. A. Charles[a] and D. Lange[a]

Abstract: *This article reviews the main novel safety features of the French N4 nuclear steam supply system, which represents the current state of the art in France. The first two units of this standardized reactor are being built at Chooz. Investigations have started in support of future designs. Only brief overviews of the Framatome safety guidelines are given.*

Production of electricity by nuclear energy has come to a crossroad in France and in the world. In France, new reactors are not expected to be built in this century, and the export market is likewise weak.

Framatome prepares the next generation of reactors by a preliminary analysis of new advanced systems and core-design concepts aiming at the optimum use of fissile materials, such as that represented by the RCVS (Spectral Shift Convertible Reactor) concept. The advanced nuclear steam supply system (NSSS) design is now represented by the N4 design, on which the two first units of the third French series of standardized reactors under construction at Chooz are based.

This article describes the novel aspects of this N4 NSSS advanced design and gives a brief insight into safety features that will govern the development of future designs.

[a]Framatome.

THE N4 ADVANCED NSSS SAFETY FEATURES

The N4 NSSS, an example of the four-loop type, is an upgraded, high-power design. This design is the result of the development of the French safety approach worked out through the design and construction of 54 units by Electricité de France, including 34 units of the three-loop 900-MW(e) class series and 20 units of the four-loop 1300-MW(e) class series.

The overall French safety approach has been extensively explained in various international meetings and documents, as, for instance, in *Nuclear Safety,*[1] and more recently at the NUCSAFE 88 meeting in Avignon, France.[2] Only novel aspects of the NSSS will be outlined.

The novel safety features of the N4 parent four-loop plant and the 1300-MW(e) plants were analyzed by the U.S. Nuclear Regulatory Commission (NRC) in 1986 (Ref. 3) and termed "a substantial improvement in safety compared to the typical U.S. four-loop PWR design for a number of potential dominant sequences." All the improvements referred to were incorporated in the N4 model together with several new features that further enhance overall safety. Table 1 lists the main N4 NSSS characteristics.

Design Features Resulting from the French Safety Approach

The French approach can be globally characterized by successive addition of *conventional*

Table 1 Main Characteristics of N4 Series Compared with P4–P′4 Series

Characteristics[a]	P4–P′4	N4
NSSS maximum thermal power, MW	3 817	4 270
Number of fuel assemblies	193	205
Reactor inlet temperature, °C	292	292
RCP flow rate, m^3/h	23 000	24 500
RCP motor power (nominal), kW	6 100	7 100
SG heat-transfer surface, m^2	6 900	7 300
SG outlet steam pressure, bar	71.0	72.9
Total live steam flow, t/h	7 780	8 650
Net electrical output (maximum), MW(e)	1 300	1 470
Mass of main components, tonnes:		
Reactor vessel	431	453
Steam generators (4)	1 744	1 620
Main NSSS components	2 900	2 820
Steam turbine	2 750	2 400

[a]NSSS, nuclear steam supply system; RCP, reactor coolant pump; SG, steam generator.

design conditions (those resulting from the three basic levels of defense in depth), *complementary design conditions* (defined as the loss of redundant safety systems when called upon), *multiple failure accident conditions* (which generated the so-called physical state approach), and finally *severe hypothetical accident conditions*. The analysis of additional conditions, performed with realistic assumptions, leads to the elaboration of emergency procedures ("H" and "U") and backup means.

Conventional Design Conditions. Conventional design conditions have been made more stringent for the N4 model in one area, the steam generator tube-rupture accident, following evidence that such rupture is not purely hypothetical. The rupture of one tube is considered a Class 3 incident with rather severe allowable consequences. The rupture of two tubes is a Class 4 accident. To avoid any risk of water discharge from the secondary safety valves, the atmospheric steam dump was improved by doubling the number of valves on each steam line [two power-operated relief valves (PORVs) and two isolation valves in series] and by qualifying the valves for water discharge. Multiple tube rupture and the coincident stuck-open secondary safety valve were also investigated. No fuel was uncovered and no borated water tanks were emptied as a result of these events, which were consequently not kept as design conditions.

Complementary Design Conditions. A major concern in the design is the progressivity of safety measures. No cliff-edge effect in design conditions and no large step in consequences may exist when considering events with a slightly lower probability than the design-basis events. Therefore a number of complementary design conditions corresponding to the total failure of redundant systems were added to the N4 list of conventional design conditions. Mitigating means, termed "backup," and procedures, termed "H," were determined to meet the French safety objectives for these complementary conditions.

The N4 safety objectives include probabilistic criteria.

—No unacceptable consequences with a frequency of more than 10^{-6} per reactor year should result from the operation of plants.

—Applied to a particular family of events, no unacceptable consequences with a frequency of more than 10^{-7} per reactor year should result from this family.

Note that formerly this objective was guidance by the Safety Authority, used only as a design condition for man-made or natural events. On the N4 project this objective was applied for the first time to justify the complementary conditions and was therefore used in the licensing process. Unacceptable consequences are interpreted here as severe core degradation, a very conservative definition that ignores the mitigating effect of the containment.

Table 2 lists the N4 complementary design conditions, and Table 3 shows the corresponding design improvements. Among these improvements is the newly implemented overpressure protection system, which allows bleeding of the reactor coolant system (RCS) in the H2 procedure (RCS bleed/feed following total loss of feedwater in the steam generator). The system was originally designed to answer questions about the reliability of the safety or relief valves after the Three Mile Island Nuclear Station accident. The system is installed on the pressurizer and comprises three discharge lines, each one equipped with a tandem of pilot-operated safety valves (Fig. 1). Each tandem is composed of a safety valve, closed at operating pressure and opened in case of overpressure, and an isolation valve in series, open at operating pressure and closed in case of failure to shut the first valve. All three lines participate in

Table 2 N4 Series Complementary Design Conditions

Condition	Procedure
Reactor trip system failure	ATWS[a]
Total loss of ultimate heat sink	H1
Total loss of feedwater in steam generator	H2
Total loss of electrical power	H3
Long-term total loss of LHSI[b] pumps, containment spray system pumps, or heat exchanger	H4

[a]Anticipated transient without scram.
[b]Low-head safety injection.

Table 3 Corresponding Design Improvements for N4 Series

Diversified anticipated transient without scram-mitigating system
Reactor coolant system bleed/feed using the pressurizer pilot-operated safety valves
Additional turbo-generator set allowing reactor coolant pressure seal injection and emergency batteries power supply
Cross-connections between safety injection system and containment spray system allowing long-term mutual backup

overpressure protection; in addition, one line ensures pressure control at a lower set pressure.

The main safety objectives are the following:

—Ensure stable operation without risk of valve chatter for any type of discharge flow condition (steam, water, or both).

—Provide capability for remote manual opening under postaccident conditions using safety-grade equipment.

—Ensure the reliability of valve reclosing (prevent the relief valve from being stuck open).

—Improve the accuracy of set-point adjustments and provide capability for periodic verification of set points and valve operability without valve dismounting.

—Maintain valve leak tightness even in case of reduced margin to the trip point.

Multiple Failure Accident Conditions. Multiple failure accidents include errors of diagnosis, use of the wrong procedure, and/or multiple concurrent systems failures beyond what had been considered in the design and which can result in severe accident situations. At such a point, it would still be possible to prevent severe core degradation. Two measures are implemented on the N4 plant.

The first measure is the development of a complete set of accident procedures based on the "physical states approach" or "symptom-oriented approach," which will replace the event-oriented procedures that are now implemented in French plants.

This approach implies (1) the diagnosis of states based on a survey of the parameters used for the different systems (primary circuit, secondary circuit, containment, and safeguard); (2) the identification of operator actions as individual objectives (e.g., residual heat removal, restoration of the water inventory, and subcriticality); and (3) the ranking of the objectives and immediate actions for each state. With such a set of procedures, the operating team should be able to avoid diagnostic errors and always perform actions that are appropriate to the cooling state of the reactor.

The second measure provides for additional capability to cope with successive failures of onsite cooling means that might occur within several days or weeks. The U3 procedure was developed to connect additional mobile pumps and a heat exchanger to restore (or increase the redundancy of) heat removal over the medium term.

Severe Hypothetical Accident Conditions. Finally, in the unlikely event that all the previously mentioned measures are insufficient, the mitigation of the consequences of severe hypothetical accident conditions (core melt) is considered. Mitigation is performed by ensuring the integrity of the third barrier to fission products, the containment. This important step in the French approach (not described here) is also based on the use of procedures and backup means, as, for instance, venting the overpressure by passage through a sand filter.[1,2]

Safety Systems Design Principles

Separation of functions is a principle implemented on the N4 and recent French plants to the greatest extent possible. It implies that one system

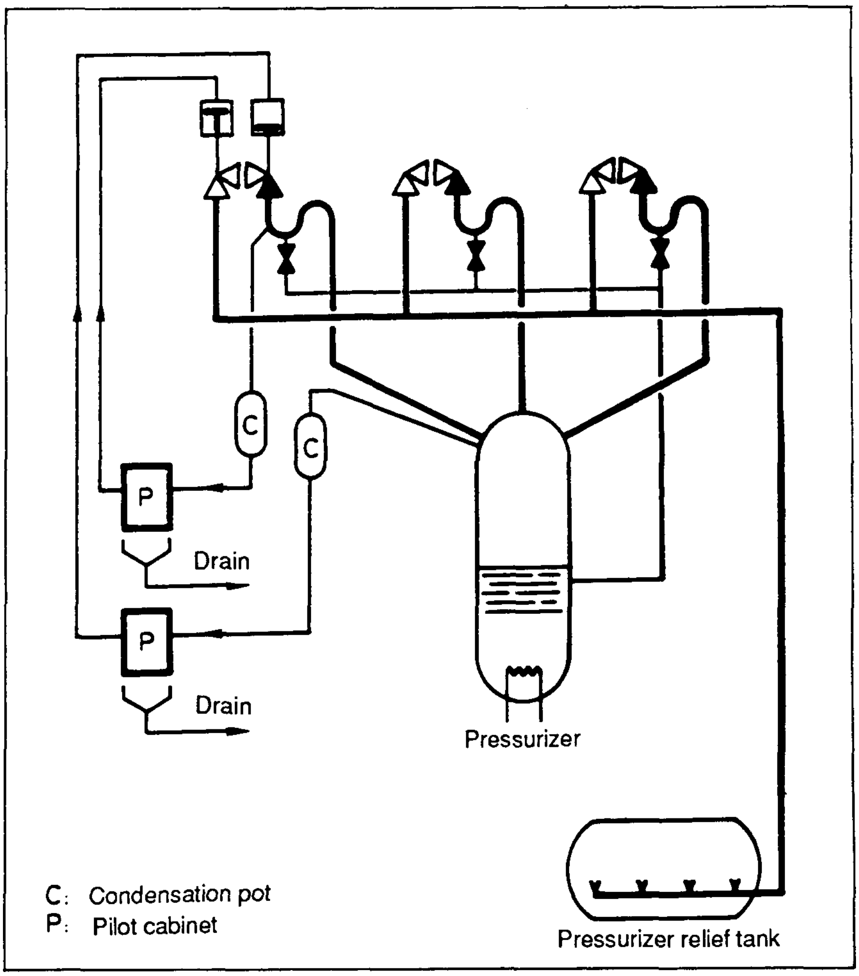


Fig. 1 Reactor coolant system overpressure protection.

is dedicated to one safety function to avoid system configuration change in case of an accident. It also simplifies plant operation and makes the system more understandable and predictable to the operator.

A general two-train system organization has been adopted as was the case for earlier French plants. The main advantage is that this system allows easy installation, easy separation of redundant paths, and lower susceptibility to common-cause failures.

In regard to provisions for maintenance, no extra redundancy is required for safety systems in standby: preventive maintenance is performed at reactor shutdown when the system is not required; if a system is unavailable (for example, as a result of a periodic test), power operation is allowed only for a limited time, following which the plant must be placed in the run-back mode in accordance with the operating Technical Specifications. According to the French operating experience, no significant plant unavailability occurs as a result of safety systems unavailability (a fraction of a day per year). Because safety systems that are used in normal operation can be maintained only while in service, such systems are required to have available either extra redundancy or a backup so that maintenance can be performed without lowering the safety level.

The complementary design conditions correspond to the total failure of a redundant system. Diversified backups are provided to face such conditions. For systems that are frequently actuated in their safety function, short-term backup is provided; for systems that are seldom actuated in their safety function, only long-term backup is provided. As far as possible, diversity is implemented on these backups to prevent additional common-cause failure modes.

Safety system characteristics are given in the Appendix.

Design Improvements Enhancing Safety

The N4 design objectives call for substantial improvements to safety characteristics. These include progress on the reliability and expected lifetime of main equipment as well as human–machine interface improvements.

A typical example of component reliability improvement is the reactor vessel. It is to be manufactured from ferritic steel with low impurity content and low initial nil ductility transition temperature (−12°C for the base metal); this will minimize the risks of fracture. Further, the shell rings are fabricated by anvil forging from hollow ingots, a process that will improve the soundness of the material near the cladding regions.

Numerous individual improvements were made early in the component design stage in the light of operating experience with 900-MW(e) series reactors to facilitate maintenance and reduce personnel exposure. They cover features to increase the lifetime of critical parts and to reduce the extent and duration of inspection and maintenance tasks, such as a reduction in the number of welds or easier access and dismantling conditions. For instance, the reactor coolant pump (Fig. 2) includes the following:

—A hydrostatic bearing conducive to longer shaft-seal lifetime.

—An oil-pressure bound-shaft coupling, which allows a significant reduction of shaft-seal inspection time.

—Main flange simplification.

In addition, for the reduction of occupational exposure, low-release materials (use of Inconel 690 for the first time on a steam generator) are being used, efficiency of the purification system is being increased, decontamination conditions are being improved, and equipment installation is being improved (e.g., location and access). Finally, two new features contribute to the improvement of the operating conditions and the operating safety: the Dispositif de Manoeuvrabilité Maximal (DMAX) core-control system and advanced control-room design.

The DMAX is a second-generation core-control system, following the Dispositif de Manoeuvrabilité Accrue (DAM) system used on previous units. Both systems use "grey" control rods. With the DAM system, the rod-control groups are moved inside the core as a "solid" set with a preset overlap between the different groups. The overlap is selected to minimize the axial power distribution offset. The operator may sometimes have to correct this offset by using the temperature-control rod group or by varying the boron concentration.

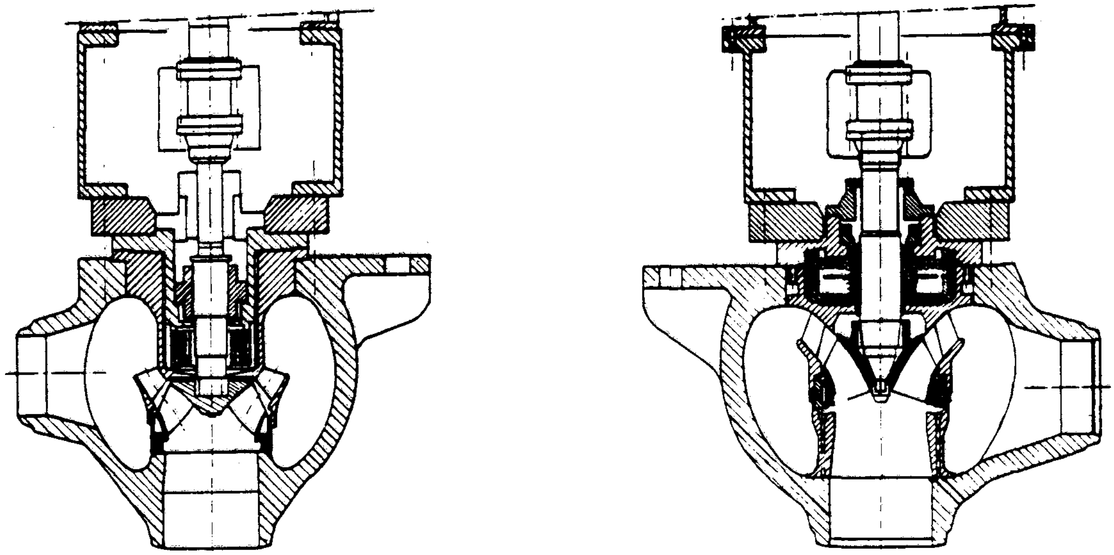
With the DMAX system, the axial offset is automatically controlled by an additional closed control loop that adjusts the overlap between the rod groups. In addition to helping the operator, DMAX achieves very efficient axial-offset control during any transients or dynamic perturbations (Fig. 3).

The advanced control room selected by Electricité de France is based on ergonomic principles aimed at facilitating the operator tasks and maximizing the understanding of events under all circumstances. Each redundant operator desk (Fig. 4) is a single working area where the operator has total access to information and control devices. The information display is presented in a highly integrated and comprehensive form.

FUTURE DESIGNS AND SAFETY TRENDS

The N4 advanced reactor, first of the French third series of standardized reactors, is now being built. It is therefore too early to define what the safety aspects of the new generation of Framatome reactors will be. N4 reactor safety is in accordance with principles laid down in the recently issued "Basic Safety Principles" of the International Nuclear Safety Advisory Group[4] and has reached an advanced technical level. It is quite obvious, then, that the goals that have directed the N4 design will remain current topics for the definition of the next generation of pressurized-water reactors.

Nevertheless, studies have been conducted on the use of passive systems and more generally on international trends in the field of nuclear safety. These trends were introduced at the NUCSAFE 88 meeting by M. J. Teillac, Haut-Commissaire à



P4-P'4 NSSSs MODEL 100
REACTOR COOLANT PUMP
23250 m³/hr
6500 kW

N4 NSSS MODEL N24
REACTOR COOLANT PUMP
24500 m³/hr
7100 kW

HIGHER HYDRAULIC EFFICIENCY	→	- NEW IMPELLER-DIFFUSER CELL - HEART-SHAPED CASING
HIGHER MECHANICAL MARGINS	→	- HYDROSTATIC SEALING AT THE IMPELLER BELT - RADIAL THERMAL BARRIER
EASIER MAINTENANCE	→	- OIL PRESSURE COUPLING REMOVAL - FLANGE SIMPLIFICATION - EASIER ACCESS TO INTERNALS

Fig. 2 N4 reactor coolant pump.

l'Énergie Nucléaire, as "concern for a further improvement in safety by use of simpler systems with more coherent design and greater reliance on passive systems" with reference to inherently safe reactors.[5]

In conclusion, recall that no type of energy generation can be inherently safe by nature. Nevertheless, the provisions worked out to extract and collect fission energy can exhibit inherently safe characteristics, such as great steam generator thermal inertia and a negative void coefficient. Inherent characteristics are obviously attractive in that they are inescapable. The designers therefore naturally rely primarily on these characteristics to ensure the safety of their reactor; they will be tempted to tailor the design to favor (or create) inherently safe characteristics. Do not forget, however, that the mission of a nuclear plant is to generate electricity economically, to be operated easily, and to ensure availability. Furthermore, just

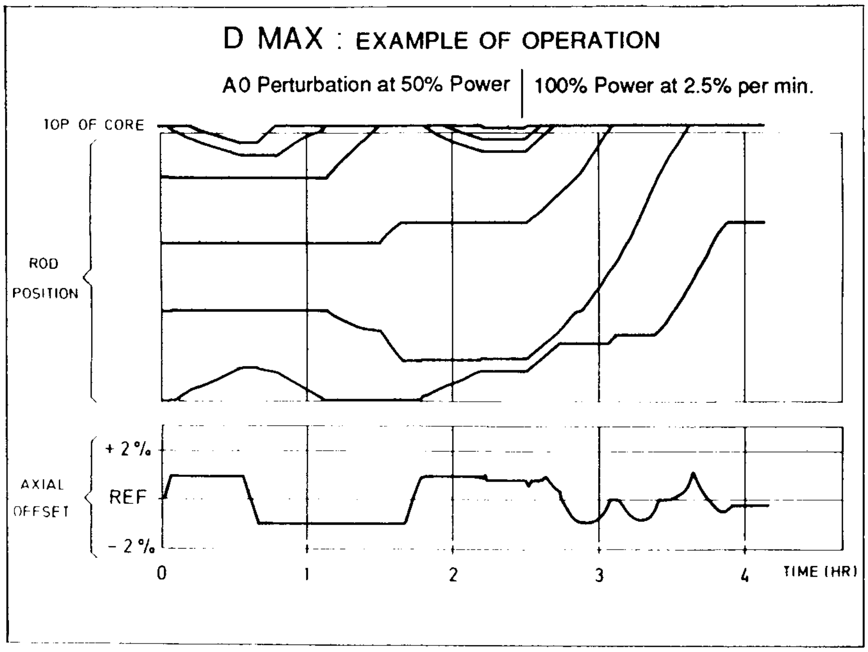


Fig. 3 Example of DMAX control.

as zero risk does not exist, most of the inherently safe characteristics will generally be attached to a physical law, a design arrangement, or a global property, all of which are valid only within a limited range of conditions and can be impaired under extreme circumstances. Therefore the notion of "inherent safety" should be viewed with great caution and is not thought to bring any actual new improvement to reactor safety.

Inherent safety characteristics cannot be enough to cope with all conceivable situations, and engineered safety systems, either active or passive, must be used to ensure the required level of safety. The present interest in passive systems relies on the fact that these systems appear to be more reliable. Because engineered safety systems are called on to operate only in case of failure, they are dependent on their actuation systems (e.g., valves, stop valves, and relays). These devices are not free of failure either on demand or by spurious actuation. Besides, even though passive systems may be more reliable, their action is necessarily less smooth and fitting than that of active systems. Therefore the type of system to be chosen depends on the nature of the elementary safety action required, whether it is a short-term action with high importance to safety, a medium-term action, or a long-term action, with or without possible human intervention.

The other trend is simplicity. The separation of functions mentioned previously as a principle for the design of French safety systems is already oriented toward operational simplicity. A simple design with a low number of passive or active elements will be more reliable and easier for operators to use. On the other hand, a simple design will be less flexible in its reactions than sophisticated systems. Here again the choice depends on the type of safety action to be performed.

This brief overview illustrates our position on the current new safety ideas. Analysis of passive systems and of basic documents such as the Electric Power Research Institute requirements, for instance, will be combined with Framatome ideas for the design of future reactors. Nevertheless, the basis of future reactor safety will remain the design and construction experience built into the N4 design and continuously enriched by operating experience.

Appendix: Main Characteristics of Safety Systems

—The residual heat removal system comprises two identical independent trains, each one having

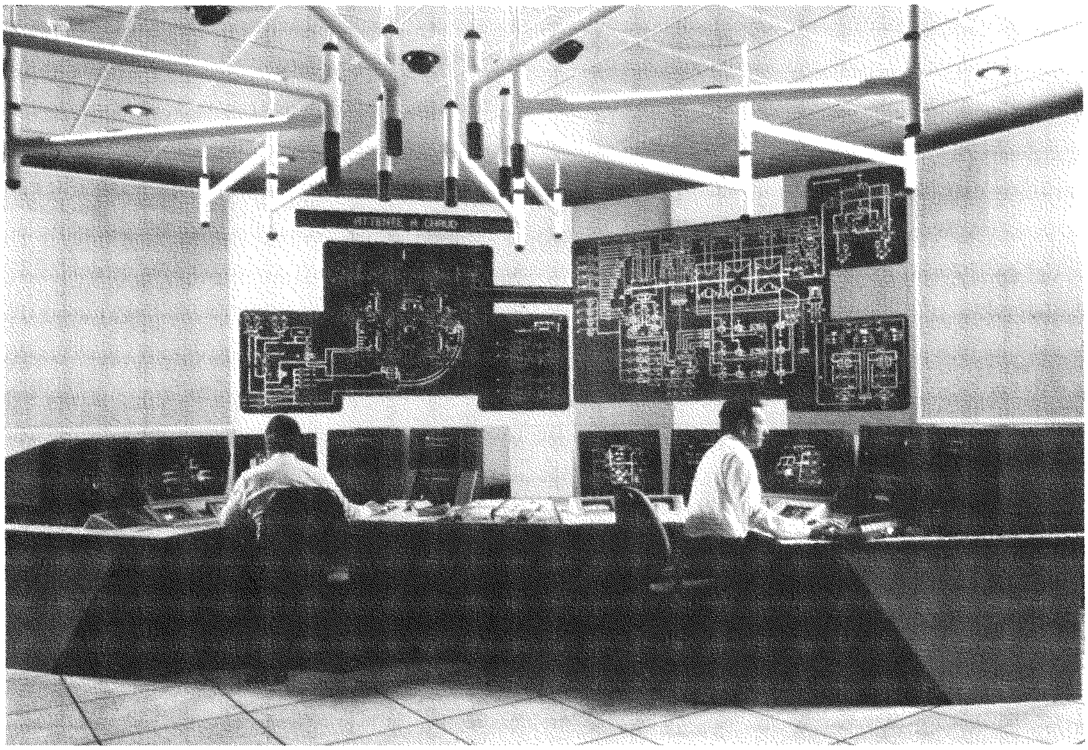


Fig. 4 **N4 advanced control-room simulator.**

one pump and one heat exchanger. It is located inside the reactor building so that, should it leak or rupture, no contaminated water would be released directly to the environment. It removes residual heat from the core during typical shutdown or after accidental conditions, once low pressure is reached.

—The safety injection system comprises two identical trains, each having one medium-head pump and one low-head pump capable of injecting into the four loops and four accumulators. It removes heat from the core to the containment in case of a loss-of-coolant accident (LOCA).

—The containment spray system comprises two identical trains, each one having one pump, one heat exchanger, and one spray ring. It removes heat from the containment to the ultimate heat sink in case of a LOCA.

—The reactor coolant system overpressure protection system, which allows the primary circuit to bleed, is composed of three lines of pilot-operated safety valves and isolation valves in series.

—The auxiliary feedwater system consists of two independent trains, each one having one motor-driven pump and one turbine-driven pump, which supply emergency water to the steam generators.

—The atmospheric steam dump allows bleeding of the secondary system. Each of the four steam lines is equipped with two power-operated relief valves and isolation valves in series.

—The reactor protection system is a fully computerized, multi-microprocessor-based system. It performs core power-distribution reconstruction and in situ computation of margins with respect to physical limits, such as departure from nucleate boiling ratio and linear power. Core power-distribution reconstruction is based on the axial–radial synthesis method; algorithms that can be rapidly computed using a microprocessor are used. Beyond accurate and nonconservative automatic protection based on safety limits, it provides data that facilitate optimal plant operation.

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Book Review of Nuclear Accidents: Intervention Levels for the Protection of the Public

By Henry B. Piper[a]

The reactor accident at Chernobyl in 1986 was unique because of the effect it had within Russia and the other Soviet Bloc countries, on neighboring countries, and, indeed, on the whole world.

The booklet reviewed here discusses radioactive releases from the accident and the intervention measures used in the non-Soviet Bloc countries to mitigate the effects of the releases. It concludes that there is a need for (1) clarification and expansion of guidance for emergency response planning and intervention criteria, (2) consistency of methods and assumptions used to develop action levels, and (3) guidance on control levels for trade involving food.

Because of the transboundary effects of the Chernobyl accident, the Nuclear Energy Agency (NEA) undertook a study of the establishment and implementation of intervention criteria for public protection. This booklet represents that study. It was prepared by the NEA Expert Group on Intervention Levels for Nuclear Emergencies (the Expert Group) and represents its views. It is published by the Organization for Economic Cooperation and Development (OECD). The Expert Group represents OECD member countries and presents no information on doses or intervention procedures in the Soviet Bloc countries.

The study approach was developed by the Expert Group to (1) review the responses and intervention criteria used by member countries, (2) search for potential means to develop generic intervention criteria, and (3) provide guidance for future development and application of such intervention criteria.

The booklet is about 100 pages long. About 60 pages are text and tables; references, a glossary, and three annexes (appendices) make up the remainder.

The Expert Group was made up of 26 participants representing 8 countries and 6 organizations. The booklet represents the views of the Expert Group and does not represent a commitment from any OECD member country. The clarity and directness of the booklet suffer from the fact that so many points of view had to be carefully expressed. For example, several paragraphs present the Expert Group as having a strong opinion on

[a]Oak Ridge National Laboratory.

one approach but points out that "some members" of the Group favored a variation of that process. However, the conclusions and recommendations of the Expert Group are quite clear, though rather general.

Emergency response planning and the intervention criteria that were in place prior to the Chernobyl event are reviewed. This discussion is centered around ICRP Publication 40, *Protection of the Public in the Event of Major Radiation Accidents: Principles for Planning,* which outlines principles for planning for protection of the public.

The radiological impact of the accident upon the member countries is reviewed and discussed. Considerable information is presented in 14 tables that show average and maximum deposition levels for cesium and iodine, individual and collective dose levels for the overall population and for specific groups, measures taken and action levels used in various countries, and broad estimates of the effectiveness of the criteria implemented. (It was interesting to discover that the levels of deposition and the doses were lower than I had expected them to be.) The spread between action levels in various countries was very large; for example, for iodine in drinking water, the action levels differed by almost a factor of 10^5 in the extreme. Estimates of the dose that was averted by protective measures were significant, and the prospect of further enhancement highlights the importance of the efforts suggested by this booklet.

It should be pointed out that the terminology "mSv" is used to mean both "mSv" and "μSv," but, with a little mental exercise, this does not present a problem in understanding the booklet.

On the basis of observations in the section on radiological impacts and the bases for planning prior to the accident, there is a discussion of the need for development and implementation of intervention criteria that are more comprehensive and consistent. The ICRP-40 approach is generally for the "vicinity" of the accident and thus is not fully appropriate for transboundary events, such as the Chernobyl accident. The discussion of this subject lays out recommendations for characterizing the accident on the basis of both spatial and temporal considerations. It also reiterates the principles of ICRP-40 and introduces the risk-based aspect of implementation of intervention measures. The recommendation that guidance be provided for transboundary trade in food products is well founded to control doses and to minimize socioeconomic impacts.

Throughout the booklet I observed that a sensible approach is being sought which would properly protect the public while preserving rational thought and action regarding international communication and commerce.

The subject, conclusions, and intentions of this booklet are so important that those of us in the nuclear community should become familiar with it and press for its recommendations to be pursued.

ANNOUNCEMENT

PROCEEDINGS PUBLISHED

In *Nuclear Safety* 30(1) we carried an article by A. Malinauskas and J. Pruett reviewing NUCSAFE 88, the International Conference on Thermal Reactor Safety, held in Avignon, France, in October 1988. The Société Française d'Énergie Nucléaire (SFEN) has announced that the proceedings of this conference is now available in six volumes for FF 850.00 and may be ordered from: SFEN, 48 rue de la Procession, 75724 Paris Cedex 15, FRANCE.

Accident Analysis

Edited by D. L. Moses

Living PRA Computer Systems

By S. C. Dinsmore[a] and H.-P. Balfanz[b]

Abstract: *This article presents a brief overview of living probabilistic risk analysis (PRA) systems and discusses some of the attributes and capabilities that have been included in the computer code SARA and in the PRA level 1 code systems IRRAS, NUPRA, PSAPACK, and SUPER-NET. The framework for the code discussions is the event and fault tree PRA methodology first systematically applied to nuclear power plants in 1975 for the* Reactor Safety Study *(NRC report WASH-1400). All code systems are under continuous development, and only the attributes fully implemented in the systems in the summer of 1988 are included.*

Probabilistic risk analysis (PRA) is based on a vast amount of logical, quantitative, and qualitative information characterizing the normal and off-normal operating modes and requirements of essentially all safety-related systems and components in a nuclear power plant. PRA is traditionally documented in a number of binders supported by many (sometimes hundreds) computer files containing the logic modes, most of the quantitative parameters, and a variety of results.

A completed PRA is a "snapshot" in time of a plant's characteristics. Any change in the plant procedures and/or hardware has the potential to change the plant's characteristics and the PRA results. The concept of living PRA requires that all such changes be evaluated and, when applicable, incorporated into the PRA. Because of the complexity of PRA and its voluminous documentation, evaluation and incorporation are generally difficult and time consuming; both require PRA experts and several person-weeks per change. Although this amount of time may be acceptable for incorporation, it is too long for evaluation.

For the support of the evaluation process in particular, various institutions in many countries have been actively involved in the development of so-called living PRA code systems.[1–7] As used here, the main features of a living PRA code system are:

—Storage of the PRA models, data, and results.

—Good, structured display of the stored information to allow traceability of any and all plant features included in the PRA.

—Features to support the systematic modification of the models and data and for the requantification of the various results.

More recently, development has begun on safety management systems[8–12] within which the living PRA is only part of a larger, more ambitious system intended to support a number of both qualitative and quantitative activities. Such activities include accessing system and component operating specifications, monitoring Technical Specification compliance, analyzing safety-related trend, and surveilling plant status.

In 1987, TÜV-Norddeutschland, in cooperation with the Technical University of Berlin, began planning such a safety management system [Safety Analysis and Information System (SAIS)] to use for nuclear power plants in its general licensing

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support activities.[12] Before the start of the current pilot project, sponsored by the Federal Ministry for the Environment, Nature Conservation, and Nuclear Safety (BMU), TÜV arranged a workshop in 1988 (Ref. 13) where various code developers and utility and government representatives discussed the potential uses of such systems; a follow-up workshop is planned for 1990. Also as part of this planning process, a detailed review of the attributes of several existing living PRA code systems was undertaken to illustrate what could be realistically expected from the living PRA portion of the SAIS. This article summarizes that review.

PRA MODEL BACKGROUND

The general PRA framework developed in WASH-1400 (Ref. 14) rests on the use of event trees to model sequences, supported by fault trees to model safety-system functions. A plant model includes these logic models and all associated failure parameters. The interface to the various thermal–hydraulic, core-melt progression, and containment response codes is found in the success criteria for the system models and the various plant state and release categories.

The plant model is structured around the potential response of a power plant to each of a number of initiating events. The global or functional response of the plant to any given initiating event is evaluated in an event tree format. Given the success or failure of each function, the event tree identifies which further functions are needed and eventually the final state of the plant. Final states are safe shutdown or various types of core damage states for level 1 PRA and various categories of radioisotope release for level 2 PRA.

The functions in the event trees can be single events but are, in general, the top events of fault trees. Each fault tree expands the failure of the function into logical combination of basic events and/or coherent inputs (i.e., transfer top events) from other systems.

Success-oriented modeling, where block diagrams or GO charts replace the fault trees, has been and is now successfully used to perform PRAs. Additionally, although most of the living PRA systems have been developed around fault trees, at least one system[11] is based on GO charts. Because of time and budget constraints, it was not possible to include consideration of success-oriented modeling in this review.

MODEL MANIPULATION OVERVIEW

A fully quantified and logically consistent PRA is assumed to exist as a baseline model. The model consists of at least three basic types of information: logical relationships, quantitative inputs, and results. Various sensitivity and modification analyses are initiated by changing the logical relationships and/or the quantitative inputs in a working model. Incorporation of changes actually implemented is assumed to be primarily an administrative decision to replace the controlled baseline study with the new model and quantitative results.

The logical relationship between an event tree, an event tree sequence, and the individual fault tree is illustrated in Fig. 1. The relationship, as illustrated, can be directly used to assemble a logic model for processing. Such fault tree linking is usually used when starting with a fault tree top event but is not always used at the event tree sequence level.

Given that the logical model has been assembled (Fig. 1), the required quantitative inputs for the end events must also be collected (Fig. 2). Once the logic model and the quantitative inputs have been assembled, the reduction and quantification of the model is performed with an appropriate computer code, such as WAM, SETS, or FTAP. Quantification begins with the basic event component types and failure parameters and ends, eventually, with the overall core damage or release frequencies.

Since a complete, baseline PRA already exists, any modification analysis will have two types of results: (1) the results at the system level and (2) the impact on the total PRA results. In the special case of updating the PRA to reflect actual plant modification, the new quantitative baseline results must be incorporated into the quantitative inputs as is also included in Fig. 2.

In general, the ability of a code system to store and subsequently link the information to support the various processes in Figs. 1 and 2 distinguishes a living PRA code system from such codes as SETS, FTAP, WAM, and many others.

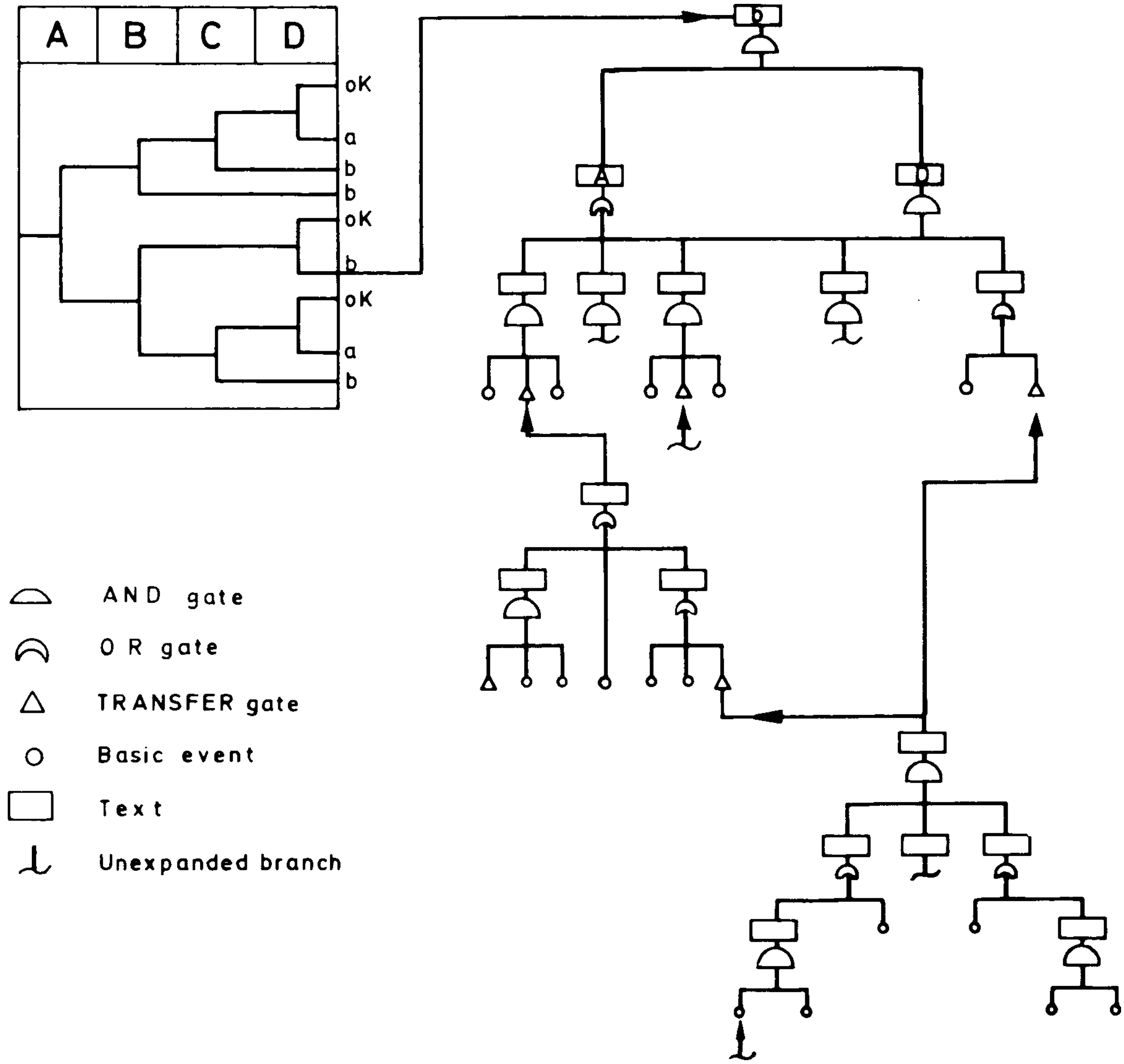

Fig. 1 Fault tree linking at the sequence level. A, B, C, and D are functions, such as scram or ECCS injection, oK, a, and b are various core damage states.

ASPECTS OF COMPUTERIZED PRA CODE SYSTEMS

When a computer system is to support the PRA as a fully integrated model, a number of issues that have been individually—and usually manually—addressed must be systematically included. A brief description of several such issues is given in the following sections.

Success Criteria

Success criteria are "hardwired" into the logic models by specifying, for example, that 2 of 4 trains are required for success. Success criteria are often initiating-event dependent [2 of 4 accumulator trains must inject for a large loss-of-coolant accident (LOCA), whereas 1 of 4 must inject for a medium LOCA] and can even be sequence dependent. Having different sequence-dependent success criteria means that it should be possible to attach function level trees at each event tree branch point.

Model Size Constraints

Because of the complexity of nuclear plants, there is a substantial amount of system interaction

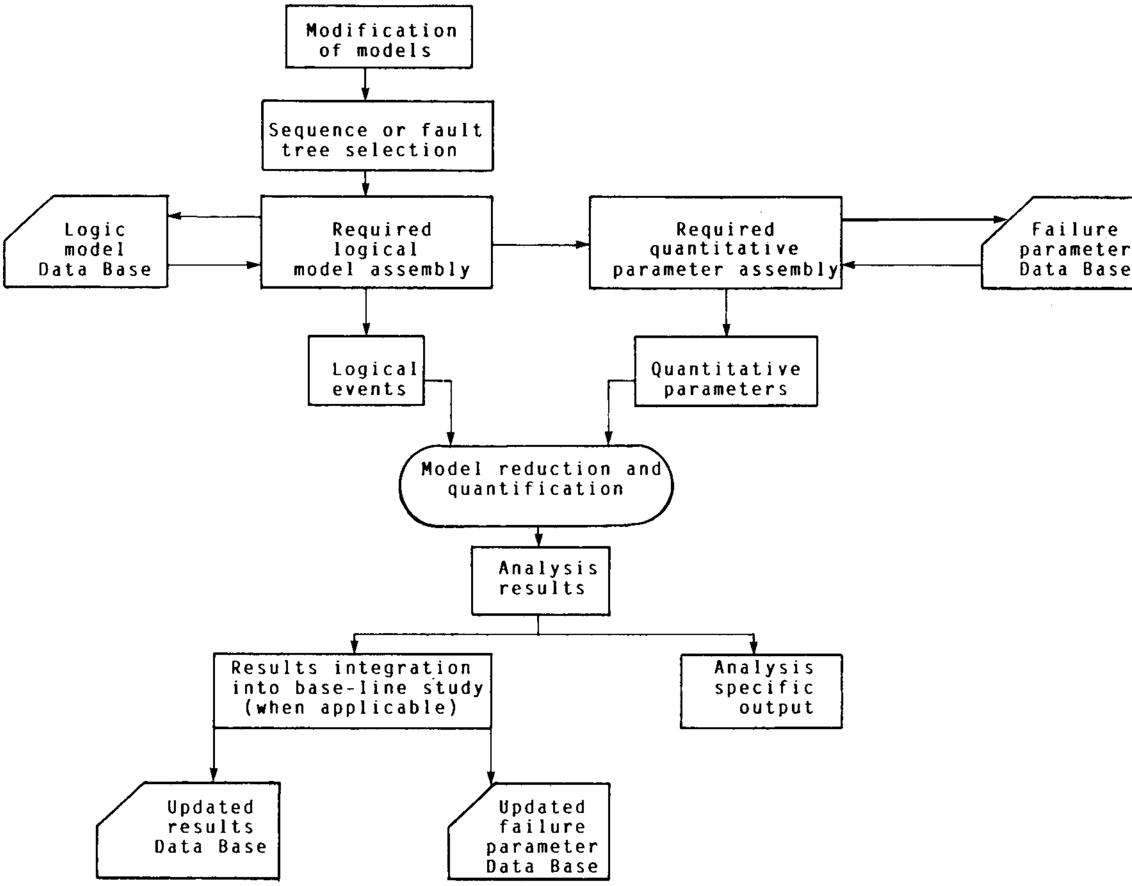

Fig. 2 Quantification input parameter assembly and output integration overview.

most clearly illustrated by transfer gates in the logical models. Consequently the resulting sequence level models are usually large and complex. Three basic techniques are currently in use to reduce and quantify sequences: (1) sequence level fault tree reduction using modularization,[15] (2) front-line system cut set equation combination,[2] and (3) event trees with boundary conditions.[15] Sequence level fault tree analysis, when performed with the use of modularization techniques, is simple and straightforward. Quite large trees can be reduced and quantified, but it is not clear if detailed, level 1 and particularly level 2 PRA sequences could always be reduced, even on workstation or mainframe-based systems.

Front-line system cut set equation combination involves first reducing the front-line system fault trees into the normal disjunctive form and subsequently combining these equations with a logical AND. With this technique it should always be possible to evaluate sequences, but care is required to ensure that low-level dependencies are not truncated in the initial reduction process.

Event trees with boundary condition analysis can break the sequences into a number of smaller pieces that are independently reduced and quantified. Although a powerful quantification technique, systematically putting these pieces back together requires the development and use of additional logical and quantitative relationships.

Sequence-to-Sequence Linkage

Transfer from one event sequence to another event sequence in another event tree is sometimes used in level 1 analysis. For example, a loss of feedwater with a stuck-open power-operated relief valve (PORV) can be transferred to the small LOCA event tree. In level 2 analysis, core damage states are (almost) always transferred to containment event trees.

Systematic transfer between event trees requires consideration of gross functional dependencies between successful functions in the source tree and the required functions in the target tree (Fig. 3). Sequences can also include consideration of previously failed functions, but this is a simplification for convenience because the normal reduction process can identify and correctly handle functional failure dependencies. Within a given event tree, both types of functional dependencies are usually included and are the reason that every sequence does not branch under every function.

Mission Time

Mission time is defined as the time after an initiating event or system demand during which a system is required to operate successfully. As with success criteria, a given function's mission time can be initiating event and/or sequence dependent. Unlike success criteria, however, mission time is not "hardwired" into the tree logic but is an additional quantitative parameter associated with a particular function or system in a particular sequence.

External Event Analysis

The analysis of external events (particularly internal fires and floods) using PRA methodology requires a substantial amount of additional failure parameters that are location instead of component dependent. The generation of these failure parameters, primarily failure probability vs. magnitude, must be performed off line. It is not clear that complete and systematic external event analysis routines should or will be included in living PRA systems. On the other hand, vital areas analysis is an important part of plant safety analysis.

Note that the relationship of component to location is component specific (not component type) and completely independent of the component-failure parameter relationship. For example, two nominally identical valves may have the same failure parameters but completely different locations. Thus vital area analysis requires additional linking and input data over and above those normally required in internal event PRAs.

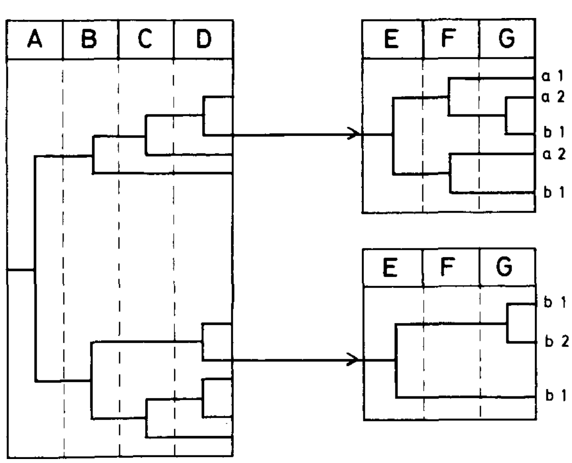

Fig. 3 Event tree functional dependencies (level 1). A, B,...G are functions, such as scram or ECCS injection, and a1, a2, b1, and b2 are core damage states.

Uncertainty Propagation

The uncertainty of interest here is the possible variation in a calculated parameter given the assumed variation in the input parameters. Evaluation of other types of uncertainties is performed with sensitivity studies using the general capabilities of the code system.

With regard to living PRA systems, the advantage would be to further propagate the previously calculated uncertainties resulting from intermediate, independent top events which have not changed and which can be quickly incorporated into modification studies.

Quality Assurance

For the application of living PRA systems in plant safety management and in the licensing and regulation process, quality assurance (QA) control is necessary. The development and maintenance of PRA models is quite complex because of the range and depth of the analyses. The situation is further complicated by the anticipated, active lifespan of the analyses—30 to 40 yr.

Two steps are involved in the continuous QA of PRA systems. First, both the original models and all subsequent modifications must correspond as well as possible with the plant; this correspondence must be documented. Second, the documented models must be used in the analysis as intended and described.

The living PRA code systems discussed in this article do not directly support the corresponding

QA, although good, graphical event and fault trees will considerably ease the review of the models by plant personnel. Safety management systems, such as the SAIS, should aim to support this corresponding QA by systematically and unambiguously linking the PRA plant model with a wide variety of qualitative function, system, and component information.

It is in the second step that the living PRA code systems can truly support the QA process. For the full support of this process, however, the code system must have a fully integrated data base; that is, any given change must only be made (and documented) in one place, and the code system must ensure that the change is reflected in all the appropriate places.

LIVING PRA CODE SURVEY

Four code systems were systematically included in this limited survey: IRRAS (Ref. 1), NUPRA (Ref. 2), PSAPACK (Ref. 3), and SUPER-NET (Ref. 4). Although not a full living PRA system as defined in this report, the SARA code[16] was also included. Crucial to the selection of these code systems was either experience with the code system by one or both of the authors, or at the least, a number of demonstrations by and discussions with the code developer. Comparison of complex computer codes is notoriously difficult and not realistically possible from published reports alone.

In addition to the experience, demonstrations, and discussions, more than 80 specific questions were developed in the project[17] and were subsequently answered by the code developers. It is emphasized that the purpose of the survey was to illustrate a range of *implemented* attributes in living PRA codes and not to make judgments on the usefulness of any particular code system. Such judgments are most properly made by individual users and must always include consideration of a number of factors not addressed in this survey, such as cost and compatibility with existing hardware and software.

For convenience, six general types of attributes are used. These are model structure, display and printing, input and connections, modification, reduction, and quantification.

Emphasis was laid on the systems' integrated model manipulation and basic quantification attributes. This "kernal" should provide integrated, plant-wide results as well as support subtasks by producing well-defined partial models such as function or sequence level cut set lists and associated failure parameters. In particular, specific details on specialty quantification routines (time-dependent unavailability, automated common-cause failure and human-error quantification, state of knowledge dependencies, etc.) were not systematically included. The applicability of many of these specialty routines depends, to a great extent, on the conditions and concerns in the various countries.

Model Structure

All the codes except SARA store the fault trees as pages generally displayed as, and accessed through, a list of fault tree pages. A fault tree page without external transfers is also a fault tree that is in itself capable of being reduced and quantified. SUPER-NET also displays and accesses the pages in an overall, plant-wide tree structure based on the transfers or on user input.

The NUPRA code supports full graphical event trees referenced by event tree name, and both NUPRA and SARA support sequence binning. PSAPACK generates sequences from a success function and dependency matrix and stores them in a list format by event tree name. SUPER-NET and IRRAS treat sequences as fault tree pages. SARA stores sequence level cut set lists only, accessed by sequence name.

All codes have the normally used gates (except SARA). IRRAS includes NOT gates and a logical TRUE/FALSE flag. All codes have a general, plant-wide basic event data base. IRRAS and NUPRA include additional fault tree page-specific data bases.

With the exception of SARA, which stores only component unavailability, a variety of failure parameters are stored, including, at a minimum, demand failure probability, failure rate, and repair time, all with associated uncertainty parameters.

Display/Printing

All systems except SARA include graphical fault trees as well as the capability to move the screen window over a tree that is larger than the screen. IRRAS always displays all labels and text, whereas SUPER-NET uses a discrete zoom option to display either labels only or both labels and text.

NUPRA and PSAPACK display only labels in the graphical tree; text display is given in a special "window" for the gate or event upon which the cursor is placed.

Basic event failure parameters, attributes, and various textual information are always displayed by positioning the cursor on the event of interest and calling up the component failure data. Provisions for entering the data file directly are also generally provided.

Graphic event trees were fully implemented only in NUPRA. In this case, the full event tree is always displayed. Function labels and text, as well as sequence labels, category, and frequency, are also displayed. Following a sequence through more than one event tree requires exiting to the control program and reentering the plant model at the appropriate place.

In general, the prints or plots of the tree are similar to the screen displays except that all codes provide for full labels and text in the printed trees.

Input/Connections

Two basic techniques are used for fault tree construction and modification. In NUPRA and PSAPACK, the logical structure is built one piece at a time with all labels entered upon creation. In IRRAS and SUPER-NET, the logical structure is first constructed and labels are automatically assigned. In general, every logical gate in the plant model should have a unique name to prevent problems from arising when linking fault trees or filling in transfers.

Basic event failure parameters can either be manually entered for each new basic event or a reference can be given to the generic data base. The reference can either be the full generic name (IRRAS, SARA, NUPRA) or a code contained in the basic event label (PSAPACK, SUPER-NET).

External transfers are special gates that refer to the top event of another fault tree. Creation of a transfer is generally done by creation of a new gate (IRRAS, NUPRA, SUPER-NET) or by "breaking" off a branch at an existing gate (IRRAS, SUPER-NET). In PSAPACK, transfers as such are not yet implemented, and any gate defined previously is filled in (i.e., copied from external files) upon creation.

The NUPRA code provides for creating and editing graphical event trees. Different logical models can be attached at each branch point in the event tree; this allows sequences to be continued in other event trees or other sequences. PSAPACK generates event sequences, whereas IRRAS and SUPER-NET treat sequences essentially as fault trees.

Modification

Baseline copy protection for sensitivity studies is normally performed by administrative control. SUPER-NET, the only code that is not personal computer (PC) based, loads a working copy of the entire plant model in main memory so that changes are not directly made to a file copy. The PC codes work with one fault tree page at a time, so the file copy (on the PC) is necessarily changed.

In all codes except SARA, the user must set up each run by defining which "top event" must be included. SARA determines which sequences must be requantified on the basis of which events were modified.

Reduction

Aside from SARA and PSAPACK, which do not include external transfers, the size of a fault tree can be somewhat controlled by replacing transfer gates with pseudo-basic events. IRRAS and SUPER-TREE provide a "switching" function between a transfer gate and a pseudo-basic event. In IRRAS, this is a global command (transfers not filled in), whereas SUPER-TREE includes control over each gate. In NUPRA, this switching must be done with the use of the normal editing functions.

All codes except SARA include automated modularization during fault tree reduction; NUPRA provides user-defined modules as well. IRRAS and SUPER-NET provide cut-off on cut set order or probability, NUPRA on probability and PSAPACK on order. In all codes, cut set files and, in all but SARA, prereduction fault tree files are accessible.

The NUPRA code combines disjunctive normal equations and PSAPACK combines cut sets at the sequence level. IRRAS and SUPER-NET work with sequence level fault trees.

Quantification

All codes except SARA provide several different component-type options (and associated basic failure parameter storage), which are usually selected by the user. SARA uses only component unavailability.

The IRRAS code requires the input of mission time for each specific analysis and uses this mission time for the entire tree. PSAPACK and NUPRA provide a place for mission time in the basic event parameters, whereas SUPER-NET does not appear to explicitly include mission time.

Cut sets are generally quantified with the rare event approximation. IRRAS, SARA, and SUPER-TREE combine cut sets in the cut set lists using the minimum cut set upper bound. PSAPACK and NUPRA combine the cut set lists with the rare event approximation. NUPRA has the option of using the third-order correction to the rare event approximation at both the cut set and the sequence cut set list levels.

The NUPRA and SARA codes, which support sequence categorization, produce summarizing information for each category. SARA also always displays new vs. old results but only at the sequence level because fault trees are not included.

All codes except PSAPACK provide several importance measures. Some of the codes appear to produce importance measures based on component attributes (i.e., component type, location, and manufacture), but insufficient information was collected to provide an overview of the possibilities.

SUMMARY AND CONCLUSIONS

The methodology to support highly integrated PRA code systems based on graphical fault trees is available. The fault tree page and transfer gate linkage has been fully implemented, including the systematic following of transfers through the various pages. Plant-wide, component failure parameter data bases have also been implemented. Model requantification, based on selection of a top event with the code system extracting the necessary details from the various data bases, has become a standard feature.

The methodology to support integrated sequence and event tree analysis is less well developed, and a variety of techniques are used or are under development. Linkages between event tree branch points and the fault trees are basically the same as transfer fault trees and are implemented. However, transfers from sequence to sequence do not seem to be fully implemented. In general, requantification of an entire PRA is still an analyst-controlled, step-by-step process at the sequence and event tree level—with the exception of SARA, which does not include the full plant model.

The current difficulties at the sequence and event tree level are certainly a result of (1) the size of the sequence level models and (2) the complexities associated with functional success dependencies.

Aside from the traditional label- and number-oriented results, presentation of summarizing information appears to be the least well developed area in living PRA systems. A variety of questions regarding results presentation and qualitative overview information were included in the checklists. More than in any other area, however, these questions were interpreted quite differently by the different developers with the general response being that all the information in the models can be extracted. Obviously, the WASH-1400 framework alone is not sufficient to systematically cover results presentation; rather, a reasonably clear idea is required of how the living PRA is to be used both within a computer-based safety management system and within the design or operating organization.

Although it is not proposed to directly equate either living PRA or safety management with degree of automation, the significant investment of manpower required to maintain and use such complex tools must be recognized. We believe that a fairly high level of automation and the eventual integration of a living PRA system within a safety management system will provide the best opportunity to effectively use these tools.

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Summary of ICAP Assessments of RELAP5/MOD2

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Abstract: *The International Code Assessment and Applications Program (ICAP) encompasses bilateral agreements between the U.S. Nuclear Regulatory Commission and 14 nations or multinational organizations. One objective of the ICAP is to assess the RELAP5/MOD2 computer code to identify its deficiencies and formulate user guidelines. To date, ICAP has assessed RELAP5/MOD2 in 20 separate studies, 10 of which have been reviewed and evaluated. As a result, three code deficiencies were identified, and four user guidelines were formulated. This article summarizes the information considered, describes the processes used, and discusses the code deficiencies and user guidelines developed therefrom.*

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Safe operation of the increasing number of light-water reactors (LWRs) worldwide requires the application of advanced thermal-hydraulic computer codes for use in studies of safety. To this end, the U.S. Nuclear Regulatory Commission (NRC) organized and coordinates the International Code Assessment and Applications Program (ICAP). The ICAP is an international cooperative program for advancing accurate methods of analyzing LWRs by (1) developing a common understanding of the ability of a code to represent important physical phenomena appropriately and supporting the quantitative determination of code accuracy; (2) sharing user experience on code assessment and compiling a well-documented assessment data base; (3) identifying code errors and inadequacies and cooperating in removing the deficiencies to maintain a single, internationally recognized code version; and (4) establishing and improving user guidelines for applying the code.

The ICAP encompasses 14 bilateral agreements between NRC and as many participating nations or multinational organizations, a list of which is given in Table 1. A common stipulation of the agreements between the participants and NRC is that NRC furnish the computer code, associated documentation, and code maintenance in exchange for member-sponsored code assessment studies. The ICAP participants have thus far conducted 20 assessment studies addressing the RELAP5/MOD2 computer code. Of these, 10 have been reviewed and evaluated.[1-10]

RELAP5/MOD2 (Ref. 11) is a best-estimate, full-system, thermal–hydraulic computer code developed at the Idaho National Engineering Laboratory (INEL) for the analyses of pressurized-water-reactor (PWR) systems. A wide variety of postulated accidents in PWR systems can be simulated with the RELAP5/MOD2 code.

A version of the code was "frozen" to ensure a consistent base for the ICAP code assessment effort. Maintaining a frozen version requires that code models be neither developed nor improved during the assessment period. The version of the RELAP5 code frozen for ICAP assessment studies is RELAP5/MOD2, Cycle 36. Responsibility for reviewing and evaluating ICAP assessments of RELAP5/MOD2 resides at INEL. Of the 20 assessment studies submitted, 10 have been reviewed.

The assessment of a large complex system code to determine the capability of the code to simulate observed phenomena accurately is not an easy task. During most reactor accident situations, major phenomena occur simultaneously and are often interrelated. For example, core thermal response during a loss-of-coolant accident (LOCA) is closely related to the system hydrodynamic response. Yet it is necessary that these and other phenomena be assessed separately if code deficiencies are to be identified and corrected. A commonly used method of assessing full-system codes is to compare code-calculated results with data obtained by controlled experiment. In general, ICAP assessments of the RELAP5/MOD2 code used this technique. The data base for each of the ten assessment studies reviewed to date is provided in Table 2. Although these bases are predominately from subscaled experiments, one base is plant data recorded during a steam generator tube rupture incident, another is critical flow data from a full-scale test facility, and yet another is primarily a void-fraction correlation. Most studies assessed multiple areas of the RELAP5/MOD2 code, as indicated in Table 2. These ICAP assessments have provided valuable information in identifying code deficiencies and in formulating user guidelines. The ICAP assessments have also identified coding errors and in some cases provided the necessary corrections. Hence ICAP provides direct feedback to the NRC-

Table 1 ICAP Participants and Organizations

Participating nations	Organization
Belgium	TRACTEBEL
Finland	Technical Research Center of Finland
France	Commissariat à l'Énergie Atomique
Federal Republic of Germany	Federal Ministry for Research and Technology Kraftwerk Union Aktiengesellschaft Gesellschaft für Reaktorsicherheit
Establishment of the European Atomic Energy Community	Joint Research Center—ISPRA, Italy
Italy	ENEA
Japan	Japan Atomic Energy Research Institute
Korea	Korea Advanced Energy Research Center
Netherlands	Netherlands Energy Research Foundation
Spain	Consejo de Seguridad Nuclear
Sweden	Swedish Nuclear Power Inspectorate Studsvik Energiteknik AB
Switzerland	Swiss Federal Institute for Reactor Research
Taiwan	Coordinating Council for North American Affairs (CCNAA)
United Kingdom	United Kingdom Atomic Energy Authority Central Electricity Generating Board Nuclear Installations Inspectorate National Nuclear Corporation British Nuclear Fuels Ltd.
United States	United States Nuclear Regulatory Commission

Table 2 Bases and Areas of ICAP Assessments of RELAP5/MOD2 (Reviewed)

Study/ reference No.	Bases	Area
1	Steam generator tube rupture	a. Steam generator liquid level incident at the DOEL-2 plant b. Vapor condensation c. Natural circulation
2	OECD–LOFT small-break (0.4%) experiment LP-SB-03	a. Critical heat flux b. Fuel thermal response
3	MARVIKEN critical flow data	a. Critical flow (subcooled, saturated steam, and low-quality two-phase)
4	FIX-II loss-of-coolant (31% split-break) experiment 3027	a. Two-phase wall friction b. Critical heat flux c. Flow regime selection process
5	Royal Institute of Technology dry-out experiments	a. Critical heat flux b. Postcritical heat flux thermal response
6	FIX-II loss-of-coolant (200% double-ended break) experiment 5061	a. System depressurization b. Critical heat flux
7	THETIS boil-down experiments	a. Core boil-off rates b. Interphase drag
8	Best-estimate vapor fraction correlation and limited test data	a. Interphase drag
9	OECD–LOFT small-break (0.4%) experiment LP-SB-03	a. Steam generator liquid b. Reflux heat transfer c. Fuel thermal response d. Vapor condensation
10	OECD–LOFT small-break (1.0%) experiment LP-SB-01	a. Critical mass flow b. Vertical stratified flow

sponsored code development activities and thereby contributes to the enhancement of the analytical capabilities of full-system codes.

On the basis of information obtained from the ten studies reviewed, three deficient areas of the RELAP5/MOD2 code were identified and four user guidelines were formulated. The deficient areas are interphase drag, critical mass flow, and critical heat flux. This article summarizes the information obtained from the assessment studies and the procedures used in the review process to identify and qualify both code deficiencies and user guidelines. Coding errors, corrections, and other suggested improvements to the code provided by ICAP participants were handled informally and are not discussed here.

IDENTIFYING CODE DEFICIENCIES

In general, specific code deficiencies are not identified by the ICAP assessment studies. The identification of code deficiencies and the formulation of user guidelines result from the review process. The review and evaluation of ICAP assessment studies at INEL are based on criteria identified in Refs. 12 and 13. Basically, the review process consists of two parts: (1) identifying potential code deficiencies and user guidelines and (2) evaluating the potential code deficiencies and user guidelines to ensure that each is fully supported by data and/or analyses and to ensure consistency with other assessment information.

The process of identifying code deficiencies begins by noting discrepancies between code-calculated results and the data base. The 10 assessment studies reviewed to date noted 28 discrepancies. These are listed in Table 3. A few discrepancies were noted by more than one assessment, whereas others were shown to be related to a common cause. For example, the discrepancy in the initial collapsed-liquid level observed in study 9 and the discrepancy in the vapor fraction observed in study 7 both result from excessive drag between the vapor and liquid phases. Thus the 28 discrepancies listed in Table 3 are not all unique.

A few noted discrepancies do not have measured data corresponding directly to the discrepancy. In some cases this information is inferred from other measured data, such as the time and location of critical heat flux (CHF). An example of this inference is provided in Fig. 1, which compares the measured heater-rod thermal response from study 4 with calculated data. These data show that the calculated time of CHF occurred ~13 s later than measured, which implies a deficiency in the CHF models. Discrepancies noted in other variables were obtained through analyses of measured data. Vapor-fraction data, for example, are obtained through the analysis of measured coolant conditions. Figure 2 compares a vapor-fraction profile calculated by RELAP5/MOD2 with that obtained through analyses of measured data from the THETIS boil-down experiments (study 7). This comparison shows significant differences between calculated and measured vapor fractions at most axial elevations, which implies a deficiency in the method for calculating vapor fractions. The maximum discrepancy is −40% at the 1.5-m elevation.

Each discrepancy noted in Table 3 is accompanied by a suggested cause. A suggested cause is the reason given by the assessment study for the discrepancy. Two studies noted discrepancies without suggesting a cause, and some assessment studies suggested more than one. The cause provided in the table, however, was considered by the review process to be the primary one. All suggested causes are potential code deficiencies. Not all the suggested causes, however, were considered for evaluation. In a few cases, code input was given as a possible cause. These cases are identified by means of a footnote in Table 3 and are not evaluated.

EVALUATING CODE DEFICIENCIES

The acceptance or rejection of a suggested cause as a code deficiency was based primarily on three considerations: (1) sufficient technical support; (2) the frequency that a particular cause was given, including consistency with other code assessments; and (3) possible relationships among causes.

Sufficient Technical Support

Several of the suggested causes listed in Table 3 were eliminated as potential code deficiencies because the data or analyses provided were considered insufficient to support the item adequately as a code deficiency. (These suggested causes are identified in Table 3 by a footnote.) For example, excessive vapor condensation was suggested by two ICAP assessment studies as the possible cause for RELAP5/MOD2 calculating pressure responses inconsistent with the measured data. One study presents evidence that RELAP5/MOD2 calculated depressurization in the steam generator secondary side when the measured pressure was increasing. These calculated and measured pressures are compared in Fig. 3. This study suggests that the discrepancy in depressurization was the result of excessive vapor condensation at the vapor–liquid interface in the steam generator dome and that the excessive condensation resulted because the code had selected the wrong flow regime and consequently the wrong heat-transfer correlation. The second assessment study presents evidence of RELAP5/MOD2 calculating the injection of emergency coolant at a faster rate than measured. The study contends that the faster injection occurred because of excessive condensation at the point of injection; this resulted in overdepressurization of the primary system. Conditions under which the excessive condensation occurred are different between the two assessments and were rejected as a code deficiency on the basis of insufficient technical support. Excessive condensation,

Table 3 ICAP Code Assessment Studies Summary of Discrepancies and Causes

Study No.	Noted discrepancy	Suggested cause
1	a. Liquid-level swell in steam generator downcomer	Excessive interphase momentum transfer (interphase drag)
	b. Steam generator pressure	Excessive vapor condensation[a]
	c. Erratic natural dryout	Form loss coefficients[b]
2	a. Time of core dryout	Core modeling limited to one dimension[c]
	b. Fuel heat-up rate	No rod-to-rod radiation[a,c]
3	a. Step increases in critical flow for saturated steam	Discontinuity in sonic velocity at phase boundaries
	b. Critical flow for saturated steam	Discharge coefficient
	c. Atypical critical flow response to input discharge coefficient	Feedback between discharge coefficient and critical-flow solution
	d. Critical mass flow for subcooled liquid	Discharge coefficient
4	a. Core pressure drop	Two-phase friction losses[a]
	b. Time of core dryout	CHF correlations
	c. System initial (steady-state) coolant inventory	Spray droplet diameter and fall
5	a. Location of CHF	CHF correlations
	b. Magnitude of CHF	CHF correlations
	c. Discontinuities in CHF	Iteration on wall temperature and critical heat flux
	d. Transition boiling	None stated
6	a. Break flow/system depressurization (system initial coolant mass)	Spray droplet fall velocity[a]
	b. Time of core dryout	CHF correlation dependence on vapor fraction
7	a. Core-dryout level response	Excessive liquid ejection
	b. Vapor fractions	Interphase drag
	c. Oscillation in vapor fraction at steady state	Periodic application of vertical–stratified flow model
8	a. Vapor fraction/two-phase density	Interphase drag
9	a. Steam generator initial collapse liquid level	Excessive steam entrainment (interphase drag)
	b. Fuel thermal response	Modeling of core[b]
	c. Accumulator injection rate	Excessive vapor condensation[a]
	d. Reflux heat transfer	None stated
10	a. Subcooled critical mass flow	Thermal nonequilibrium effect in discharge nozzle
	b. Sudden draining of upper-plenum and hot-leg liquid	Application of vertical–stratified flow model

[a]Insufficient information and/or analyses to support as code deficiency.
[b]Input related.
[c]Beyond the intended capability of the code.

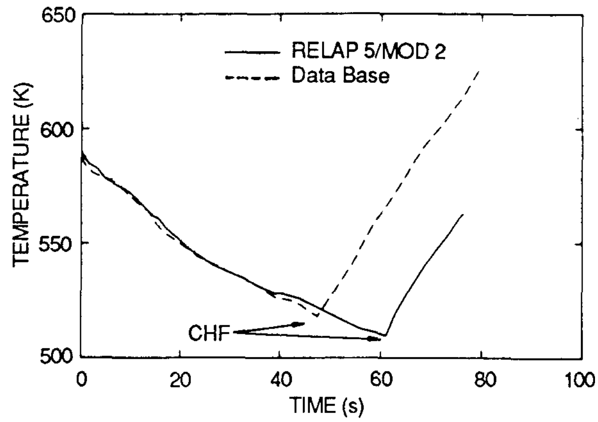

Fig. 1 Time of critical heat flux inferred from cladding temperature data.

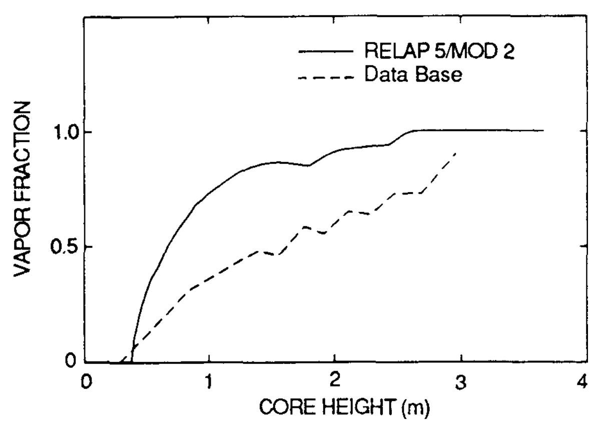

Fig. 2 Vapor-fraction data inferred from coolant conditions.

however, was included as a possible area of deficiency to be investigated.

Frequency

The number of references made to a particular cause was also a consideration in qualifying suggested causes as code deficiencies. Multiple references suggest a greater chance that a deficiency actually exists. For example, CHF correlations were cited three times as the cause for discrepancies observed in fuel or heater-rod thermal responses. Although the evaluation of each occasion indicated that additional information would be needed to support CHF fully as a code deficiency, the RELAP5/MOD2 CHF correlations and models were identified as a deficient area of the code. This decision was based primarily on the number of times CHF was given as a cause, but consistency with other code assessments was also a consideration.

Several ICAP assessments of the RELAP5/MOD2 code are based on LOCA (pipe breaks). So that break flows that were consistent with the experiment could be obtained, it was necessary for some assessments either to input or recommend the input of discharge coefficients other than 1.0. The input of discharge coefficients to RELAP5/MOD2 modified the flow area of the simulated break by a factor equal to the coefficient. Thus discharge coefficients other than 1.0 are not representative of the true break area but rather are indicative of possible deficiencies in the break-flow models. A discharge coefficient less than 1.0, however, can be justified as a method of compensating for two-dimensional effects, such as vena contracta, that are not simulated by the one-dimensional RELAP5/MOD2 code. A discharge coefficient more than 1.0, however, represents a nonphysical situation, and its required use provides strong evidence of deficiencies in the break-flow models. Consequently ICAP assessment studies citing the discharge coefficient as the cause were combined with those citing critical-flow models, and the RELAP5/MOD2 critical-flow models and correlations were identified as an area of code deficiency.

Relationships

The evaluation of suggested causes for inclusion as code deficiencies was also accomplished by establishing relationships between causes. For example, it was possible to link the excessive liquid ejection cause from assessment study 7 to interphase drag. One ICAP assessment study presented data showing that RELAP5/MOD2 had calculated initial heater-rod dryout at an intermediate core elevation rather than at the top of the core, as observed during the test, and suggested that this resulted from excessive liquid ejection. A similar but independent assessment[14] of the RELAP5/MOD2 code presented evidence of calculated collapsed liquid levels in the core that were lower than those determined from measurements. The independent assessor, believing these lower levels were related to excessive liquid ejection caused by excessive drag between the vapor and liquid phases, replaced the interphase drag correlation for the bubbly/slug flow regime with a slightly modified version of the correlation considered by

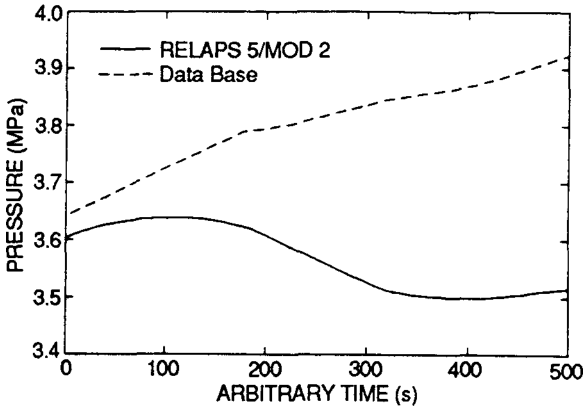

Fig. 3 Pressure responses in steam generator secondary.

Bestion[15] in his work with the French CATHARE code. The results with this new correlation were excellent. On the basis of this evidence, we concluded that excessive liquid ejection was related to and actually caused by excessive interphase drag.

In addition, two ICAP assessment studies suggested that, when the vertical–stratified flow model was applied by the RELAP5/MOD2 code, discrepancies related to interphase drag were observed. One study noted oscillations in the calculated vapor fraction at steady state which were traced to the periodic switching on and off of the vertical–stratified flow model. Another study calculated draining of liquid from components in excess of that observed in a test and noted that the movement of liquid occurred as the vertical–stratified flow model was invoked. When the vertical–stratified flow model is invoked by RELAP5/MOD2, the interphase drag calculation is altered. Hence invoking the vertical–stratified flow model can result in a significant and sudden change in the interphase drag. Consequently the two vertical–stratified flow causes were combined with and included as interphase drag causes. On the basis of these established relationships, the RELAP5/MOD2 interphase drag models were declared a deficient area of the code.

As a result of 10 ICAP assessment studies and the identification and evaluation process for defining code deficiencies, 3 areas of the RELAP5/MOD2 code were declared to contain known deficiencies. These are listed in Table 4 and consist of the interphase drag models, the critical mass flow models, and the critical heat flux models.

Table 4 also includes two areas of the RELAP5/MOD2 code that we believe may contain deficiencies. These are excessive vapor condensation at vapor–liquid interfaces and excessive fall velocity of liquid droplets in a vapor environment. Although both were identified in two separate ICAP assessment studies, neither was included as a known code deficiency because we judged the information contained in the studies to be insufficient.

USER GUIDELINES

Another important part of reviewing and evaluating ICAP code assessment studies is the extraction of user guidelines. User guidelines are recommendations intended to improve code performance. A primary criterion was imposed for qualifying user guidelines (i.e., the guideline must be

Table 4 RELAP5/MOD2 Code Deficiencies and User Guidelines

Code Deficiencies

Identified Areas of Deficiency:

1. Interphase drag
2. Critical mass flow
3. Critical heat flux

Areas of Possible Deficiency:

1. Excessive vapor condensation at vapor–liquid interfaces
2. Excessive droplet fall velocity in vapor environments

RELAP5/MOD2 User Guidelines

Confirmed User Guidelines:

1. No benefit is realized by explicitly modeling discharge piping or nozzles with length-to-diameter ratios less than 4 ($L/D < 4$).
2. Modeling of loop-pipe connections to the reactor vessel should use the option for cross-flow connections.
3. Two radial nodes in thin fuel or heater rod cladding (thickness $\leq$3 mm) produce acceptable temperature distributions (<0.5 K difference with 2 vs. 10 nodes).
4. Essentially the same boil-off rates were obtained with 6 vs. 12 axial fluid cells. Additional cells should be considered for system pressures less than 4 MPa.

Nonconclusive User Guidelines:

1. An acceptable, more efficient, steady-state calculation may be obtained by either relaxing the convergence criteria or by using the transient option.
2. Discharge coefficients less than 1.0 may be necessary when calculating the critical flow for saturated steam.

supported by calculated results). Figure 4 is an example of a supporting calculation. The calculated data are from assessment 7, which included a course axial node representation of the rod bundle. The comparison data in Fig. 4 show that essentially the same core dry-out responses are obtained with 6 fluid cells (axial nodes) as with 24.

User guidelines are also evaluated to determine whether restrictions in application are needed. For example, the data in Fig. 4 support using 6 vs. 24 axial fluid cells; however, differences in core-dryout responses between the two models were observed to increase with decreasing pressure. The system pressure for the data in Fig. 4 was 4 MPa. Hence a guideline based on these data needs to be restricted to system pressures ≧4 MPa.

User guidelines extracted from the 10 ICAP assessment studies reviewed to date are listed in Table 4. In general, user guidelines are provided without discussion of the evaluation process. Two categories of user guidelines are provided: (1) confirmed user guidelines supported by calculation and (2) user guidelines that have not been confirmed. The latter are listed in Table 4 as "nonconclusive guidelines." Category 1 guidelines include any restrictions deemed advisable in their application.

SUMMARY AND FUTURE WORK

Since the release of RELAP5/MOD2, Cycle 36, as the "frozen" version for ICAP assessment work, numerous ICAP assessment studies of the code have been conducted and submitted to NRC. Of those submitted, 10 have received review and evaluation for the purpose of identifying code deficiencies and formulating user guidelines. As a result, 3 areas of the RELAP5/MOD2 code were found deficient, and 4 user guidelines were defined. Code deficiencies and user guidelines are summarized in Table 4.

Resolving these code deficiencies will influence decisions regarding safety issues. One primary safety issue confronting LWRs is the integrity of fuel-rod cladding during accidents. The three areas of identified deficiency do affect calculated cladding temperatures and thus will influence judgments made regarding cladding integrity during many accidents. Consequently these areas of identified deficiency will be given special attention during the development of the next version of the RELAP5 code (i.e., RELAP5/MOD3). Because RELAP5/MOD2 will form a large portion of the basic RELAP5/MOD3 code, an effort has been initiated to evaluate and revise the RELAP5/MOD2 calculational models and correlations associated with these code deficiencies. Several ICAP members and NRC are now cosponsoring development of the RELAP5/MOD3 code. As a result of this cooperative effort, RELAP5/MOD3 will be an improved thermal-hydraulic computer code capable of more accurate simulations of postulated accidents, and this will provide an enhanced safety analysis tool for evaluating PWR systems.

The ICAP assessments of the RELAP5 series of codes will continue, as will the review and evaluation of those assessments. The ICAP code assessment activities also contribute to the information required for the quantification of code uncertainty.

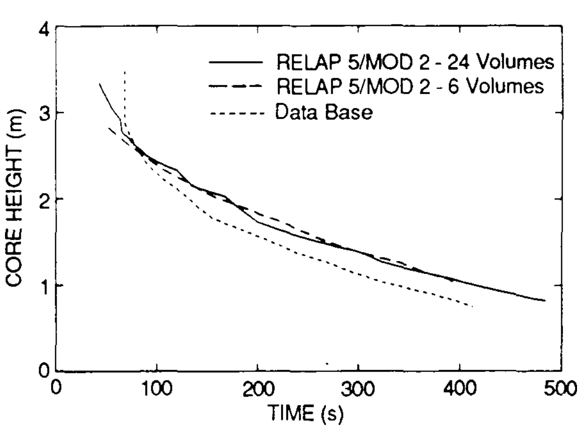

Fig. 4 Core dry-out responses.

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CEC SEMINAR ON METHODS AND CODES FOR ASSESSING THE OFF-SITE CONSEQUENCES OF NUCLEAR ACCIDENTS

Athens, Greece, May 7–11, 1990

The Commission of the European Communities (CEC) is planning to hold this seminar with the cooperation of the Kernforschungszentrum Karlsruhe (German Federal Republic), The National Radiological Protection Board (UK), and the "Demokritos" National Research Centre for Physical Sciences (Greece).

The seminar will include both invited and contributed papers and will focus on

— presenting, evaluating, and analyzing the results obtained from the CEC MARIA Program (Methods for Assessing the Radiological Impact of Accidents)
— describing the probabilistic accident consequence code system COSYMA developed within the framework of the MARIA Program
— reviewing and discussing similar work being undertaken elsewhere, especially outside the European Community
— providing an opportunity for constructive exchange of views between experts on the state of the art of methods for accident consequence assessment

For additional information, contact Dr. G. N. Kelly, Scientific Secretariat, Methods and Codes for Assessing the Off-Site Consequences of Nuclear Accidents, CEC, DG XII.D.3, ARTS 2/51, rue de la Loi, 200, B-1049, Brussels, BELGIUM. Telephone: (32) 2-235-6484. Telefax: (32) 2-236-2006.

Control and Instrumentation

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Thermal Performance Monitoring System at Maanshan Nuclear Power Plant

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L.-Y. Liao,[a] and Y.-B. Chen[a]

Abstract: *The objective of this project is to develop an on-line computerized system that can do nuclear power-plant turbine-cycle performance monitoring and analysis during normal operation and recommend remedial methods after finding thermal efficiency degradation or missing power problems. A prototype system was developed and implemented at Taipower's Maanshan Nuclear Power Station (a pressurized-water reactor) in 1987. This system can be used to identify the malfunction of plant sensors, provide concise thermal performance information, and easily familiarize the plant operators and engineers with the simple and user-friendly features of the turbine cycle.*

The Institute of Nuclear Energy Research (INER), in cooperation with Taiwan Power Company (TPC), engaged in the full-scale American Society of Mechanical Engineers (ASME) performance tests for three nuclear power plants [including both boiling-water-reactor (BWR) and pressurized-water-reactor (PWR) units] from 1978 to 1984. All test instrumentations were set up by the INER, and all calculation techniques and test procedures were based on the ASME PTC-6 code.[1,2] During those tests, the engineers of INER developed an on-line monitoring and analysis system for the thermal performance of the turbine cycle and the arrangement of the turbine cycle components. Because the thermal kits of a BWR power plant are quite different from those of a PWR power plant, two monitoring systems were developed for the Kuosheng power plant (BWR) and the Maanshan power plant (PWR). This article will describe the Maanshan thermal performance monitoring (MSTPM) system only.

The MSTPM system puts plant on-line test data through a central computer to calculate performance indexes and presents the operation performance information and diagnostic messages of each turbine cycle component. During normal operation, this system can be used to monitor either Unit 1 or Unit 2 of the Maanshan power plant, depending on the operator's choice.

SYSTEM DESCRIPTION

The structure of the MSTPM system established at the Maanshan station consists of two major parts—Data Acquisition System (DAS) and Monitoring and Analysis System (MAS)—as shown in Fig. 1. Four Tektronix series color graphics terminals and two color graphics copiers are used at different locations to display turbine cycle performance based on the operator's request during normal operation. A detailed configuration of the system set up at INER and Taipower Headquarters is also shown in Fig. 1. The data are communicated among these stations through telephone lines. The distance between Maanshan and TPC is about 500 km, and the distance between

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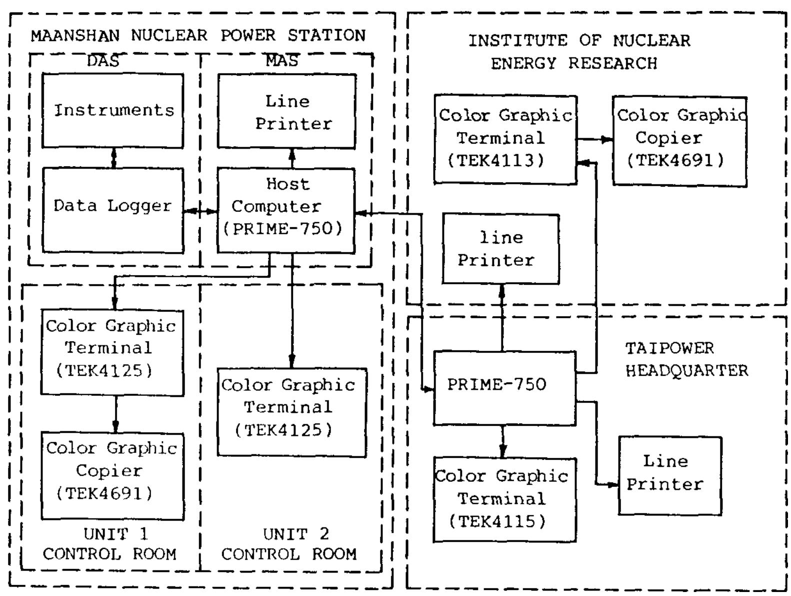

Fig. 1 Maanshan nuclear power plant thermal performance on-line monitoring system structure configuration.

TPC and INER is about 50 km. Engineers can update the source program and monitor plant performance on each terminal.

The major functions of DAS are to process signals, including performance parameters that can be used to provide turbine cycle diagnostic information, and to show the trends of all plant data. Detailed descriptions of the thermal performance calculations in MAS are:

1. Set up baseline data of each point after a refueling outage. The baseline data include reference data and their allowable variation range at different thermal power levels.
2. Compute the difference between the baseline data and actual operating performance data. If the test data deviate from the allowable variation range, they will be replaced by the baseline data.
3. Calculate the pipe leakage flow rate and enthalpy of the moisture separator and reheaters (MSR) on the basis of the mass and energy balances.
4. Calculate the steam flow rate and enthalpy of the turbine extraction line and monitor the dumping valves status of the heater drain.
5. Check condenser vacuum by monitoring the coolant water inlet temperature and heat load.
6. Calculate the heat rate, used energy end point, high-pressure and low-pressure turbine efficiencies,[3] and other important performance parameters on the basis of the heat balance of the overall turbine cycle.
7. Compare the corrected heat rate with the reference heat rate and diagnose the plant operating conditions.
8. Display the turbine cycle performance flow diagram and diagnostic messages on the color graphics terminals and print concise output from the line printer.
9. Update the diagnostic messages as plant performance engineers increase their operating experience.
10. Set up the monthly trends of the plant data, which are printed by the line printer and displayed on color terminals.

The preceding ten steps of MAS can be executed by simply pressing function keys on the keyboard of the Tektronix color graphics terminal as shown in Table 1. When the performance engineers press the function keys to activate the monitoring system, the evaluation results will be shown on a turbine cycle flow diagram, and the diagnostic messages will be displayed on an individual component diagram (see Fig. 2) on the color graphics screen. A total of 23 color figures are used in the MSTPM system to display detailed

Table 1 MSPMA Function Key Operation Procedure

Items	Pushbutton: Function key	Pushbutton: Selection	Function descriptions
Performance evaluations	F1		On-line real-time evaluations
	F2		Routine acquisition for daily operation
	F3		Plant instrument data performance trend
Components display	F4	1 to 6	Feedwater heater No. 1 to No. 6 display
	F5	1	High-pressure turbine display
		2	Low-pressure turbine display
		3	Feedwater pump turbine display
	F6	1	Condenser display
		2	Condensate pump display
	F7	1 to 2	Moisture separator and reheaters display
	F8	1	Steam generator display
		2	Steam packing exhauster display
		3	Feedwater pump display
	S1		Turbine expansion line display
	S2		Important performance trend display
	S3	1	Test cycle performance flow diagram
		2	Reference cycle performance flow diagram
		3	Design cycle performance flow diagram
	S4		Return to test cycle performance flow diagram
	S5		Erase flash mark and dialog area
	S6		Erase screen
Data-base process	S7	1	Baseline update
		2	Diagnostic message file update
File output	S8	1 to 6	Performance results and analyses output
	Hard copy		Color graphics hard copy

thermal performance information for individual turbine cycle components. The important thermal performance parameters, such as temperature, pressure, enthalpy, and flow rate, are shown at the inlet and outlet locations of each component figure with different colors. The data with deviations exceeding the allowable range will be indicated by a flashed square. The possible causes, the occurrence frequency, and the last date of occurrence will be analyzed and shown on the screen. The capability of this system to make data trends provides useful information to detect a slow degradation of the system performance after the last refueling outage. In addition, comparisons of trends of two identical heater trains and four identical MSR trains are also provided to identify the causes of performance degradation.

DIAGNOSIS METHOD

After a few years of operation, turbine cycle thermal performance could be degraded because of abnormal operation, instrument damage, component performance degradation, etc. So that the possible degradation can be traced, a diagnosis package was developed in the monitoring system. Furthermore, the heat-rate compensation, usually in accordance with one of the heat-rate codes,[4,5] is calculated by an experienced performance engineer. The diagnosis flow diagram shown in Fig. 3 indicates that the process includes the validation of test data and diagnosis of the plant efficiency. In Fig. 2, the data in the parentheses are test data, and the data located above the parentheses are input data. If the difference between the test data and the reference data is

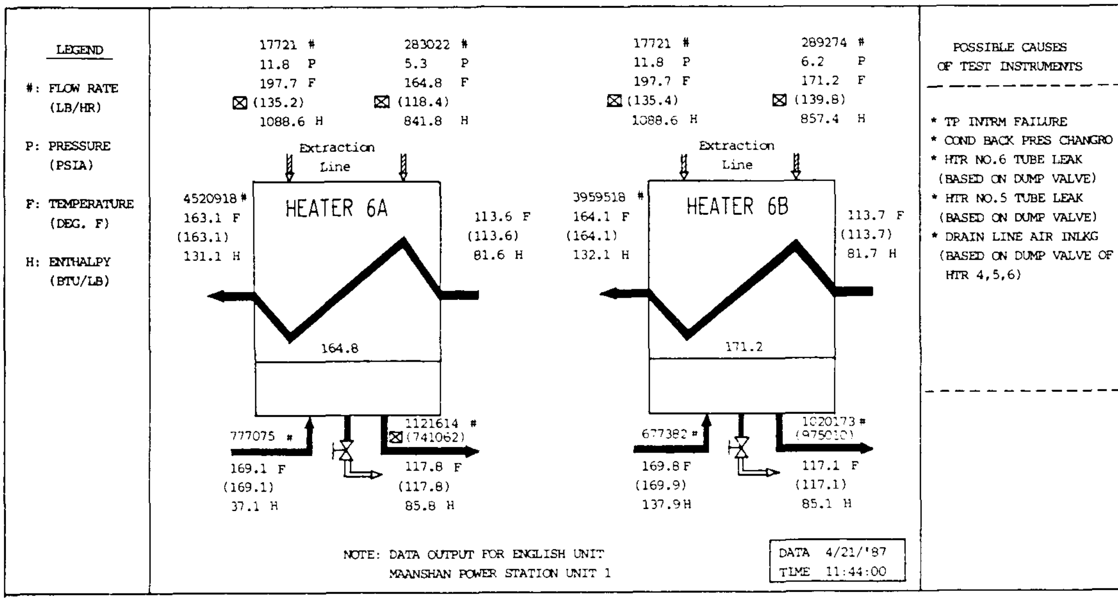

Fig. 2 Heater No. 6 performance diagram.

within the allowable variation range, then the test data can be used as input data. Otherwise, the input data should be replaced by the reference data, and a flashed red square mark will appear at the left side of the parentheses. When the plant efficiency is poorer than the reference efficiency, a yellow square frame will flash at the display locations of the heat rate and the components that deviate from normal operating conditions on the turbine cycle performance diagram.

The test data considered to be important in the determination of the heat rate are initial (throttle) pressure, final feedwater temperature, and low-pressure turbine exhaust pressure.[6] The initial moisture at turbine inlet should be included also in the determination of the heat rate. However, the measurement of the initial moisture is very difficult during normal operation. Therefore the test initial moisture in MAS is taken from design data. On the basis of the daily statistics of plant operation, the allowable variation range of the initial pressure, the final feedwater temperature, and the exhaust pressure was determined to be 10 psia, 2°F, and 0.4 in., respectively.

If the deviations of the test data from the reference data are within the allowable variation range, a corrected heat rate is calculated on the basis of the correction factors provided by the manufacturer. If the deviation of the preceding three test data exceeds the allowable variation range, the correspondent component is considered to be in abnormal condition, and the heat rate is calculated on the basis of the reference data. The plant efficiency diagnosis is based on a comparison of the calculated heat rate with the reference heat rate. If the calculated heat rate is almost equal to or less than the reference heat rate, the plant is considered to be in normal condition. Under the normal condition, if a flashed red square mark appears on the screen, it could be caused by the malfunction of the sensor. If the plant is in an abnormal condition, a flashed square frame will appear at the display locations of the heat rate and the abnormal components. The system will provide diagnosis messages and the difference of heat rate from the abnormal component to the performance engineers for operation reference.

RESULTS AND DISCUSSION

A prototype thermal performance monitoring system was developed and implemented at Taipower's Maanshan nuclear power station in 1987 and is now being tested at the plant site. The MSTPM system provides turbine cycle performance information to plant engineers whenever the

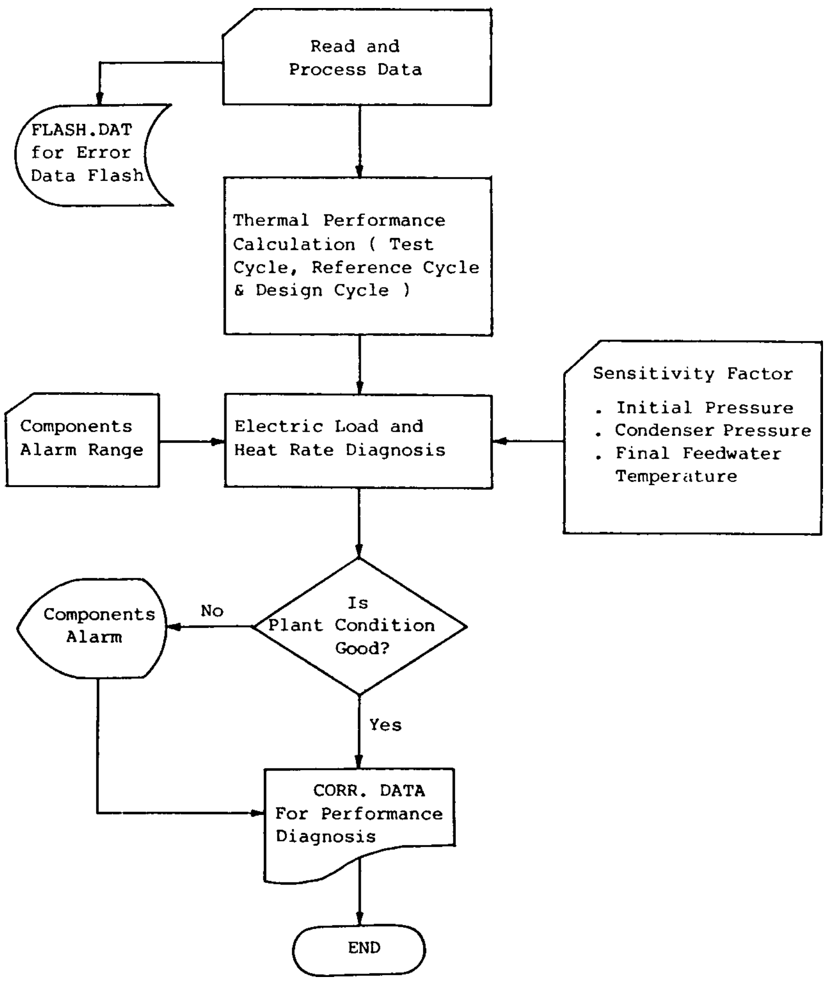

Fig. 3 Thermal performance diagnosis flow diagram.

function keys are pressed during normal plant operation. The performance information includes turbine cycle performance diagrams, component performance diagrams, turbine expansion lines, test data trends, important temperature distribution diagrams, and analysis and diagnosis tables. So far, the major objectives achieved by the system are:

1. To identify quickly the malfunction of plant sensors.

2. To provide concise thermal performance information (with a few pages), which greatly reduces the hand-calculation load for performance engineers.

3. To easily familiarize the plant operators and engineers with the turbine cycle because of its simple and user-friendly features.

Note that, if the component of the turbine cycle is changed or repaired, the cycle performance will be altered. Therefore the baseline data and allowable variation range should be updated by the experienced engineers to fit the new turbine cycle. The reliability of the diagnosis messages of the MSTPM system can be increased by the increasing experience of the performance engineers.

It is expected that, after a few years of testing, plant performance engineers, with the assistance of the developers, can prioritize the items in the diagnostic checklist according to the frequency of the abnormal occurrence. With the possible causes in hand, the performance engineers can make a decision whether to correct the malfunction of the sensor immediately or to arrange a proper maintenance schedule according to the results of a cost–benefit analysis.[7] In the future, an expert system based on knowledge from experience can also be added to enhance greatly the capability of the system.

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Design Features

Edited by D. B. Trauger

Warning Systems for Nuclear Power Plant Emergencies

By J. H. Sorensen[a] and D. S. Mileti[b]

Abstract: *Over 200 studies of warning systems and warning response were reviewed. The major findings are as follows: First, variations in the nature and content of warnings have a large impact on whether or not the warning is heeded by the public. Relevant factors include the warning source; warning channel; the consistency, credibility, accuracy, and understandability of the message; and the warning frequency. Second, characteristics of the population receiving the warning affect warning response. These include social characteristics, such as gender, ethnicity, and age; social setting characteristics, such as stage of life or family context; psychological characteristics, such as fatalism or risk perception; and knowledge characteristics, such as experience or training. Third, many myths about public response to emergency warning now exist and are at odds with knowledge derived from field investigations. Some of these myths include the "keep-it-simple" notion, the "cry wolf" syndrome, public panic and hysteria, and public willingness to respond to warnings. Finally, different methods of warning the public are not equally effective at providing an alert and notification in different physical and social settings. Most systems can provide a warning given three or more hours of available warning time. Special systems, such as tone-alert radios, are needed to provide rapid warning.*

Most people involved with nuclear safety view alert and notification systems as a physical means of communicating with the public about an emergency. The options for doing so are succinctly defined in the Federal Emergency Management Agency (FEMA) guide[1] and in other reviews.[2,3] Recent Atomic Safety Licensing Board rulings, such as in the case of Shearon Harris Nuclear Power Plant, have defined state-of-the-art warning systems for nuclear plants in terms of technology.

Indeed, the nuclear industry is a leader in implementing state-of-the-art warning systems technology. Systems are now required to alert and notify people in a 10-mile radius of the plant in 15 min and ensure essentially 100% notification within 5 miles. However, the choice among alternative communications hardware is only one aspect of building effective warning capability. The purpose of this article is to discuss some neglected aspects of a state-of-the-art public warning system. First, we define a warning system in broader terms than an alert/notification system. Second, we discuss public response to warnings by way of identifying some common myths about how people respond to emergency warnings. Third, we examine the links between plants and offsite organization. Fourth, we discuss the style and content of effective warning messages. Fifth, we review different means of alerting and notifying the public. Finally, we discuss the importance of monitoring public response to an emergency.

REDEFINITION OF A WARNING SYSTEM

A warning system is broader in scope than an alert/notification system. The alert/notification

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system consists of a means of providing an audible or visual signal that emergency conditions exist (alert) and then communicating an instructional message (notification). A warning system has three basic components: a detection subsystem, an emergency management subsystem, and a public response subsystem.[4,5] The first stage in the decision-making process is the detection of hazard or the recognition that the technology poses a hazard. This stage is chiefly the responsibility of the plant, although at least one state is moving to have independent monitoring of plant data. Once the hazard is detected, the second key decision is whether or not the hazard poses a threat to safety. Once the threat is judged to be significant, the detector/assessor must decide whether or not to alert the public or officials of the hazard and potential damages and then determine who should be notified of the threat. The notification of a public official typically results in the activation of an emergency response system. The organization initially notified must decide who else to involve in a decision to warn. Once mobilized, emergency managers must decide whether the risks warrant a warning or protective action. Finally, they must decide what type of protective action is needed and whether or how to warn the public. This process is often interactive with numerous dynamic communication flows regardless of the scale and complexity. As such, the model implicitly recognizes the need for integration between the subcomponents, the need for timely and effective communication linkages, and the importance of decisions, including those associated with public response.

The warning-response process begins when the warning is heard.[6] Hearing a warning is often insufficient by itself to make people take action. The next stage is understanding the warning. Then people must come to believe that the warning is true and accurate.[7] Next, people must personalize the message to make it relevant to themselves. Finally, they must decide to take action and overcome constraints to taking that course of action. Throughout the process, a variety of factors influence hearing, understanding, believing, personalizing, deciding, and behaving. A major one is the process of confirmation,[8] which depends on both the nature of the warning effort and the characteristics of the receiver. Although much of this knowledge has come from the study of natural disasters, it is also applicable to technological hazards[9] and nuclear power-plant emergencies.[10,11] In addition, public education is part of the public alert and notification process in that it primes people to understand what to do when a warning situation occurs. Several summaries of the warning process from a sociological perspective[12–20] and on the warning experience at Three Mile Island Nuclear Station[21–30] are available.

The basic point when viewing a warning system in this manner is that its chief function is to maximize public safety. The suggestions in this article can improve the implementation of alert/notification systems and can allow officials to take fuller advantage of the large investments made in alert/notification technologies.

MYTHS OF WARNING: HOW THE PUBLIC RESPONDS TO WARNINGS

Many emergency managers believe in a set of popular myths and perceptions about warnings and public response to warnings that exist in the United States and that all too often constrain the effectiveness of warning systems when implemented. These myths include (1) people panic during disasters, (2) warnings should be short, (3) false alarms and the "cry wolf" syndrome are real problems, (4) people respond best to a single spokesperson, (5) people will act when they hear the first warning, (6) people will blindly follow instructions, and (7) people can remember siren signal patterns. In designing and implementing a warning system, officials and decision makers should not fall prey to these myths. They are dispelled as follows:

First, the public simply does not panic in response to warnings of impending disasters except in two, somewhat predictable circumstances. The first is in screenplays developed by Hollywood and Tokyo filmmakers—these fictions are likely the reasons for the existence of the panic myth in the first place. The second is during situations in which there is closed physical space, such as a situation with an immediate and clear source of death when obviously not everyone will be able to escape.[31,32] Panic does not follow a warning except in these very rare circumstances. Note that panic behavior is different from elevated stress, which is a psychological response to warnings that the public and media often label as panic. The downside of the myth of panic is that warning officials are reluctant to tell the truth or may withhold information because they are afraid of causing panic.

Second, the public rarely, if ever, gets too much emergency information in a warning.[33] It is true that people do not remember all the information contained in a warning if they hear it only once. Detailed messages can and should be repeated in an emergency. Emergency warnings are simply not subject to the 30-s rule known to operate in Madison Avenue attempts to sell toothpaste and deodorant soap. People are "information hungry" in a warning situation, and they should be provided with all the information they need. This myth is often reflected by the terse or glib message protocols that are designed to guide information dissemination in an emergency.

Third, the effectiveness of people's responses to warnings is not always diminished by what has come to be labeled the "cry wolf" syndrome. Two issues regarding false alarms are significant. The first concerns a false alarm that leads to public response, such as an evacuation. In this case, if the bases and reasons for the "miss" are told to the public and understood, the integrity of the system will be preserved.[34] The second concerns repeated activation of the alert mechanisms. If such false alarms occur and no attempt is made to explain why they were false alarms, subsequent public response to the alert of an event could be affected negatively.[35] This is particularly true of inadvertent sounding of sirens if such malfunctions are frequent and not explained. Eventually, the public can be expected to ignore the sirens in a true emergency. However, if false alarms are explained, they can actually enhance public hazard awareness and ability to process risk information during later warning events. As such, many false alarms are better viewed as opportunities than as problems. A good emergency plan will have a procedure for explaining false alarms. Decision makers should also be assured that the public prefers to err on the side of caution.

Fourth, people at risk who are the targets of emergency warnings want information from a variety of sources rather than from a single spokesperson.[36] This procedure helps people (1) confirm the warning information and the situation and (2) believe the content of the warning message. However, different warning messages from multiple spokespersons are not desirable. Instead, the objective could be achieved in one of two ways. Different spokespersons could all deliver the same message, or a panel of spokespersons could deliver a warning a multiple set of times.

Fifth, people simply do not take action in response to warning messages as soon as they hear the first warning.[37] Instead, people seek more information from people they know and other information sources about the impending risk, the situation, and the response. People call friends, relatives, and neighbors to find out what they plan to do, and they also turn on radio and television to get more information. They wait for a second, third, or fourth official warning before responding. A good warning plan should call for frequent messages in the early stages of emergencies.

Sixth, people will not blindly follow instructions in a warning message unless the basis for the instruction is given in the message and that basis makes common sense.[38] If instructions in an official warning do not make sense, people typically will behave according to other information sources that do make sense. Warning messages should clearly define the rationale for all recommended actions.

Last, people do not typically remember what various siren signal patterns mean but may try to find out the reason for a siren sounding if it continues or is repeated. Sirens, therefore, are best viewed and used as calls for the public to seek out other emergency information rather than as signals that should elicit adaptive protective actions by the public. Therefore it is inappropriate to use, for example, a steady tone to indicate an evacuation and an undulating tone to indicate sheltering. An exception may be the reported and frequent use of siren drills to which response has become automatic; this use is largely inappropriate for the general public but may be of use in work settings or in special situations that can be supported by an intensive education program.

Fear of public panic in response to warnings, the idea that warnings must be so short as to rob the public of needed information, fear of false alarms based on the presumption that the "cry wolf" syndrome is a law of nature that cannot be reversed, and the other myths just reviewed have acted in previous emergencies as constraints to warning systems achieving their general goal, that of helping to maximize the probability that the public at risk will respond in ways to minimize the impact of a disaster. It is hoped that dispelling these myths in the planning process will improve warnings in an actual emergency.

ESTABLISHING COMMUNICATION NETWORKS

The link between nuclear plants and the people in the community is not direct. Warning decisions usually are made by state, city, or county officials. The first step of a warning is the offsite notification of the appropriate community officials or point of contact. This step can be accomplished in a number of ways. Conventional communication systems, such as telephone and radio, are not viewed by experts as highly reliable forms of communication. Telephones can fail (sometimes from the same event that caused the accident) or may be busy. Radios often operate at different frequencies, are inoperable, or are difficult to use because of heavy traffic on the appropriate frequency. As a result, communication systems are designed to overcome such problems. These include dedicated telephone lines (separate lines not linked with commercial traffic), dedicated radios, pagers, and special alarm systems. Such are standard practices in the nuclear industry. New technologies are being developed that can provide even greater reliability. These include fiber-optic networks to exchange data, satellite communications, and microwave radios.

Good communication extends beyond hardware. One of the better predictors of good communications in an emergency is the quality of interpersonal interactions during nonemergency times. People who know each other will work together in a crisis.[6] Another is knowing who will be communicating with whom during an emergency as well as what they will be communicating about. One way to promote good communication is to conduct practice exercises. Another way is to get people in the system to know one another. A social event, such as a picnic, is one mechanism to improve emergency communications.

WRITING WARNING MESSAGES

A well-constructed message prototype for an emergency is important to the quick dissemination of information. The system and content of a message can have a dramatic effect on public response. Enough research has been conducted to discern a poor message from a good one and then even a good one from one that reflects state-of-the-art practices.

One of the clearest and most consistent of the conclusions from research is that the warning message itself—what is said in terms of both substance and style—may be the most important factor in determining the effectiveness of a warning system. The content and style of the actual warning message, among a few other factors, shape the extent to which an endangered public engages in protective action.

Warning messages that seek to maximize the style that enhances adaptive public response, as well as include the content of certain items that enhance public response, have a greater chance of being successful than the ones that ignore all or some of these warning message attributes. In the sections that follow, we review the elements of both style and content that should be considered in writing a public warning message. Before reviewing these sections, however, we will consider what we expect will be a "knee-jerk" response to what follows.

Most people have the impression that public warning messages must be short or the endangered public will become confused or lose interest in the subject. People do have a short attention span, and it is true that messages to sell toothpaste, deodorant soap, and other products are best kept short. But major collective emergencies like nuclear power-plant accidents are different from advertisements in terms of how willing a public is to listen to information. Emergency warnings of impending catastrophes convert an information-adverse public (you only have 30 s to convince me that your toothpaste will make me more sexy) into a public that more closely approximates an information-hungry public (why are we at risk, do you really mean me, how long do I have, what is it you think I should do, etc.). Warning messages that honor the basic "keep it short" principle that creates successful marketing campaigns are grossly inappropriate in public emergencies. Short, inadequate messages set a diverse at-risk public on an information scavenger hunt to fill the information void. Short warning messages are more than inadequate; they can and have been dangerous because they can, in information-stingy warning circumstances, lead people to friends, neighbors, relatives, superstitions, biases, and a raft of other "information providers" to fill the information void. These other sources may provide inaccurate information that creates rumors, ill-conceived situational risk perceptions, and subsequently incorrect public response or lack of protective actions. The sections that follow address the style and content of public emergency warning messages that are appropriate to include in all efforts to plan for and use public warnings. Figure 1 provides the framework for reviewing message protocols. Although we discuss only the elements of style and content, a state-of-the-art message should stand the test of all 25 cells of this 5 × 5 matrix.

Warning Content

Five specific topics are important to include in assembling the actual content of a public warning message. These topics are hazard or risk, location, guidance, time, and source.

Hazard. A warning message must provide the public with information about the impending hazard that has precipitated the emergency warning. Every warning should consist of two parts: a description of the event that may occur and an explanation of how it is a risk to people. For example, it is inadequate for a warning to state that radiation may soon escape from a nuclear power plant; instead, the warning should describe how the radiation will filter into the air like a cloud and then travel with the wind while becoming less and less concentrated. This example is not a prototype for nuclear power-plant radiation release messages; instead, we simply hope to point out that the hazard for which warnings are issued must not be left in a "black box." The warning must describe the character of the impending hazard: a building of pressure may cause an explosion that will allow radioactive particles to be released from a small hole in the reactor. If a hazard is well described, people are better able to understand the logic of protective actions: close the car windows while evacuating because the risk is in the air, get out of the streets because it is safer in a building basement, and so on. The general point to be followed in reference to describing a hazard in a warning is that it should be described in enough detail that all members of the public understand the physical character of the disaster agent from which they are to protect themselves. Vagueness in warning messages will result in different members of the public defining the hazard in different ways and then in responding in ways consistent with those different definitions. Informing the public about the physical characteristics of the hazard in warning messages will minimize the likelihood of members of an endangered public misperceiving the hazard and subsequently making wrong decisions about what to do.

Guidance. The content of an emergency public warning message must also include information about what people should do about the impending hazard. A warning must provide the public with guidance about how to maximize their safety in the face of impending disaster. It cannot be assumed that members of the public will know what would constitute an appropriate protective action. The protective action must be described. On the surface, this point may seem too obvious; actually, it is not. Warnings, for example, must do more than tell people in danger that they should evacuate. Evacuate for some may be to the front yard. Instead, the process should be defined; for example, "evacuate on Route 6 until you are in Friendly Township."

Location. Warnings must also include the location of the impending hazard. Risk or hazard and location are closely linked. Warnings must detail the location of who is at risk and who is not at risk, and they should do so in ways readily understood by those who are to receive the warning. For example, a warning could say, "The area of town that will be affected will be between Second and Fifth Streets from Elm Avenue to Magnolia Boulevard." If there is reason to be concerned about the perceptions of other residents who are safe, then the warning should address them; for example, "People who live in other parts of the city will not experience any danger," and the warning should then explain why.

Time. The content of public warnings must also address the timing of public response. It is important to inform members of the public who are the targets of warnings about how much time there is for them to act and engage in protective actions before impact and how much time there is before they might have to initiate protective actions. For example, "the plant conditions will not be serious before 10:00 p.m. this evening, but to be

MESSAGE STYLE	MESSAGE CONTENT				
	Hazard	Location	Guidance	Time	Sources
Specificity					
Consistency					
Accuracy					
Certainty					
Clarity					

Fig. 1 Elements of a warning message.

on the safe side, you should be past the eastern border of the county line by 9:45 p.m."

Source. The final dimension of warning content is the source of the warning. The source should be identified in the warning and be as much a component of it as is information about risk, guidance, location, and time. Sources of warning are best able to enhance believability and appropriate response if they are from a mixed panel (scientists, officials, and familiar persons); for example, "the mayor and the head of civil defense have just conferred with scientists from our local university and the Nuclear Regulatory Commission, as well as with the head of our local Red Cross chapter, and we now wish to warn you that. . . ."

Warning Style

Each of the five parts of warning content (hazard, location, guidance, time, and source) is readily juxtaposed against the five dimensions of warning style (specificity, consistency, accuracy, certainty, and clarity). By doing so, it is possible to consider the soundness of twenty-five separate elements of a warning message (see Fig. 1).

Specificity. The style of a warning message is best if it can be specific regarding the area at risk, the actions that people should take, the character of the hazard, and how much time people have to complete protective actions. Obviously, on many occasions, specificity on all these content items cannot be high because they are unknown or only known imprecisely. Even on these occasions, however, the warning message itself need not be nonspecific. Furthermore, the style with which it is written must remain specific. For example, "we do not know nor can it be known which buildings in the city are the safest, but we do know that most everyone will be safest if they stay inside and do not attempt to evacuate."

Consistency. The style of warning messages must also be one of consistency, both within messages and across different messages. Inconsistencies can exist within a message for a variety of reasons and in many different ways. For example, it is inconsistent to tell the public that a nuclear power-plant accident may result in a release of radiation but that they should not worry. It is inconsistent to tell the public not to worry about a potentially hazardous event. Inconsistencies across different warnings are numerous in most emergencies as more is learned about an impending hazard and as updates are issued to the public. Inconsistencies can appear, for example, as new information reveals that the actual character of the hazard has decreased or increased, the number of people at risk has become larger or smaller, and so on. In these circumstances, which often occur, consistency can be rendered across messages and warnings by simply repeating what was last said, what has changed, and why.

Certainty. Third, the style of a warning message is best if it contains certainty about the factors about which it is conveying information. When there are low probabilities or ambiguities associated with a hazard's impact, the message should be stated with certainty (even about the ambiguity). For example, "there is no way for us to know if there really is going to be an explosion in the reactor, but we have decided to recommend that the emergency planning zone be evacuated now and that all should act as if the explosion is a real threat." Certainty in warning messages, however, extends beyond actual message content and also includes the style of delivery. The warning message should be spoken by the orator delivering it as if he/she believes or is certain about what is being said. This concept is, perhaps, best conveyed by means of this analogy: recall being a small child and recollect your mother saying, "don't do that or you'll get a spanking," and then recall how her delivery of the message helped you know whether or not a spanking would actually follow.

Clarity. Clarity, a fourth style attribute of warning messages, simply means that warnings must be worded clearly in simple language that can be understood. For example, "a possible transient excursion of the reactor resulting in a sudden relocation of the core materials outside the containment vessel" would better be clarified by stating, "radiation will be released from the nuclear reactor."

Accuracy. The fifth and last style attribute of warning messages is accuracy. A warning message must contain timely, accurate, and complete data. If people learn or suspect that they are not receiving the whole truth, they may well lose the ability to believe a message or hold its sources as credible. Accuracy is enhanced simply by being open and honest with the public.

CHOOSING THE CHANNEL APPROPRIATE FOR THE PUBLIC

A distinction exists between public warning systems that provide an alert, provide a notification, or provide both an alert and a notification. The alert function of a warning is a signal that something out of the ordinary or unusual is occurring that requires people to seek more information. Notification is the process by which people are provided a warning message and information. A combined system serves both purposes. Examples of alert technologies include sirens or alarms. Examples of notification technologies include emergency broadcast systems, radio and television, and cable override. Examples of dual systems include tone alert radios, telephone dialing systems, loudspeakers, and public address systems. Some systems, depending on how they are used, may not fall into precise categories. A helicopter equipped with loudspeakers is a dual system, but in reality it typically does not provide notification because people cannot hear the broadcast message.

Alert Technologies

Sirens/Alarms. Although technology can provide an audible signal to most populations at risk, it may be expensive to implement. These audible warning devices are designed to alert the potentially threatened population very rapidly. A few types of sirens have public address (PA) capabilities as well, but most only emit a noise. Siren systems are limited by their lack of instructional messages. At best, they tell people to seek further information unless an intensive program of public education instructs people what to do when the signal sounds. Such a program may be useful when the same response is desired every time a warning is issued. Different signals, such as a wavering signal vs. short blasts, are rarely differentiated by the public. Relying on different warning signals is not recommended. Other problems that constrain the use of sirens and alarms are false alarms caused by technical failures, equipment failures in emergencies, maintenance problems, coverage problems (particularly in adverse weather), difficulties in propagating sound into buildings, and public indifference to sirens. Despite all these problems, siren systems are a main component of many warning systems in use today. In fact, all commercial reactors except the Edwin I. Hatch Nuclear Plant use sirens as part of their warning systems.

Modulated Power Lines. Existing electrical power distribution technology enables specialized warning systems that use power-line modulations to activate an alert system. When the system cycle-per-second frequency is altered, devices linked to electrical circuits can be activated to turn on a warning light, a buzzer, or a siren. Many of the advantages of tone-alert systems hold for this type of warning device. Modulated power-line technology, however, is relatively expensive to install, test, and maintain. In addition, it cannot be used if electrical systems fail. It is now in use at one nuclear plant.

Aircraft. In special cases airplanes and helicopters can be used as part of the warning process. Sirens or bullhorns can be carried by low-flying aircraft to provide an alert or warning message. In addition, they can drop prepared leaflets containing a warning message. This type of warning channel is useful in reaching remote populations or populations that cannot be reached by normal communication channels. Disadvantages include difficulty in obtaining access to aircraft, maintenance, and high cost. A further problem is obtaining sound systems that can broadcast a message that can be heard over the noise of the aircraft itself. No nuclear plant now uses aircraft, although some utilities are studying it more closely as a method that might be used to issue warnings to users of remote recreational areas.

Notification Technologies

Radio. Radio is often a major method for disseminating warning information because it can quickly reach a large number of people during nonsleeping hours. Certain radio stations have been designated emergency broadcast stations as part of NAWAS (national warning system). These stations usually have arrangements with local civil defense offices or other government agencies to broadcast emergency warnings for most hazardous situations. Other radio stations usually broadcast warnings as well. The use of radio as a warning method will continue to be a major practice in emergencies. Prearranged plans for notification and the use of standardized messages often accelerate the speed with which a warning can be issued over the radio. One disadvantage of the radio is that a broad area is often covered by the broadcast, including areas not at risk. Second, all information must be conveyed verbally; thus the use of graphic materials will be excluded. Third, radio reaches only a small portion of the population at night. Fourth, because stations are privately operated, problems regarding the priority of warning broadcasts can arise; however, such problems can be largely prevented by means of formal agreements and exercises. Most utilities have agreements with Emergency Broadcast System stations to broadcast warnings.

Television. Warnings are also broadcast over commercial television by interrupting normal programming or displaying scrolled text on the bottom of the screen. Television reaches a large number of people, particularly in the evening. Like radio, it is a poor method to use during sleeping hours. Television is a particularly good method for issuing warnings about slowly developing events. It is likely to take longer to issue a warning over television stations except when prewritten scrolled messages are used. One major advantage of television is the ability to use graphic information, such as maps or diagrams, in the warning. Like radio, many utilities use television stations to broadcast emergency information.

Cable Override. In many urban areas, people watch cable television, which means that local stations play a smaller role in reaching the public. As a result, systems have been developed to broadcast a scrolled or broadcast message over all cable channels. Thus a person in Cheyenne, Wyo., watching a Chicago station or a movie channel would still receive a tornado warning. The override systems are usually operated by local civil defense offices in coordination with a cable television station; prearranged conditions and agreements on the use of such systems are required. The same advantages and disadvantages of normal television apply.

Dual Technologies

Personal Notification. Personal notification involves having emergency personnel go door to door or to groups of people to deliver a warning message. This type of warning can be used in sparsely populated areas; in areas with a large seasonal or diurnal population, such as a recreation area; or in areas that are not covered by electronic warning capabilities. The chief advantage of personal contact is that people are more willing to respond to a warning because they are more likely to believe that a danger exists. The disadvantages are that (1) it is time-consuming to implement this method and (2) it may require the commitment of many vehicles and personnel.

Loudspeakers/PA Systems. It is feasible to use existing public-address (PA) systems to notify people in areas that are covered by such systems, including various institutional populations or commercial establishments. Often schools, hospitals, prisons, nursing homes, sports arenas, theaters, or shopping centers have PA systems. In addition, portable loudspeakers on vehicles can be used to warn nearby populations. They are often used in conjunction with personal notification procedures. Existing PA systems supplement other warning system communication networks. They are useful in reaching small segments of the population in confined settings. For a PA system to be effective, a link that ensures quick and accurate message dissemination is needed. Without a good communications link to the operators, PA systems are not highly useful.

Portable loudspeakers increase the speed of warning populations without other means to receive the warning. They are particularly useful at night when many people are asleep. Their chief disadvantages are that (1) it is often difficult for people to hear a warning broadcast from a moving vehicle and (2) it is difficult for them to confirm the warning, particularly if they only heard part of it.

The combination of door to door with portable loudspeakers is called "route alert." This practice is common as a backup warning system or as a primary warning system in sparsely populated areas.

Tone Alert Radio/Pagers. Tone alert radios are specialized warning devices that can be remotely activated. They provide a warning signal, and some types can then broadcast a verbal warning message. Such a radio operates in a standby condition. Upon receiving a code, it emits a tone and broadcasts a prerecorded or read message. The code and message are broadcast from a radio transmitter that typically has a range of 40 miles. The radio receiver operates on normal electrical power; furthermore, some have battery backups. The advantages of the tone-alert systems include a quick dissemination time, the combination of an alerting signal with specialized messages, and their around-the-clock availability. Disadvantages include maintenance problems, misuse, misplacement, unavailability during power failures, limited broadcast range, and difficulty in using outdoors. Some utilities are beginning to adopt tone alerts as a backup for indoor notification, particularly in a 5-mile radius.

Telephone/Automatic Dialers. Two types of automated dialers now exist: switching and computerized dialing equipment. These systems could be potentially used to reach a large number of people in a relatively short time. Switching technology has recently been developed which is capable of simultaneously calling hundreds to thousands of exchanges using automatic switching equipment. Some systems will automatically hang up phones in use and block out incoming calls during the transmission of the emergency message. These systems make use of existing private-party phone lines and telephones. Most of the modifications and special equipment are installed at the phone company. These systems play prerecorded messages that can be updated fairly quickly or broadcast messages that provide timely information.

The chief advantage of telephone warning systems is the ability to quickly disseminate a message to people at home. Automatic dialing systems, however, are limited in their use by high cost. It is unclear what fraction of a large local phone system can be simultaneously contacted. Another problem to consider is that people who are not near a phone will not receive a message. Because of these issues, automatic telephone systems are now chiefly used to disseminate warnings within an interorganizational network, such as emergency response teams or institutional facilities at risk. Recent developments make this option attractive for small communities or for areas of a community where a prompt warning is needed. Automated ring-down systems are used in many locations to notify institutions in the emergency planning zone. No switching-equipment based systems exist at commercial power plants, although one has been installed at a nuclear facility in Gore, Okla.

Performance of Warning Technologies

On the basis of data from 14 historical warning situations, we can evaluate the performance of warning systems.[7,39–44] Apparently, in most events that are detected early enough to provide a lead time of a minimum of 3 to 4 h, at least 90 to 100% of the population can be warned without the use of a highly specialized warning system.[45] The warning systems used for these events can be described as ad hoc. They involve a combination of emergency efforts, including door-to-door notification by law enforcement personnel, driving through affected areas using portable loudspeakers and sirens on emergency vehicles, and disseminating warnings over radio and television stations, including emergency broadcast stations. Permanent sirens and other, more sophisticated warning technologies were not used to warn the public about these events.

People have been warned about these events by a mix of three message sources: emergency officials, such as police officers or emergency workers, who go door to door or through the streets with loudspeakers; informal sources, such as friends, neighbors, or relatives, who make personal or telephone contact; and the mass electronic media, such as radio or television. The mix varies among events, but the reasons for variations in the mix are not well understood.

One factor that differentiates the mix is available warning time (Figs. 2 and 3). For events with only a short amount of warning time, the warning comes mainly from local officials and informal contacts. Often these events occur at night when people are not tuned to the media. The media play a more important role in short-fused events that occur during the day or the evenings. In addition, the media play a significant role in events with long lead times. In such situations, officials do not provide the initial warning but may personally notify people in high-risk areas.

The data also suggest that informal warning is likely to occur in an emergency. For events that have a very short lead time, it appears that a general rule of thumb is that, for almost every household that receives a warning from an official, another household is notified informally before officials can provide the warning. In one case, 74% received a warning from an informal source.[7] In more diffuse situations, the role of informal warning diminishes. Thus the actual timing of the warning dissemination is greatly accelerated by social processes that seem to occur in the course of most disasters. During longer events, warning information is exchanged informally by many of the people at risk, even though the first notification is not from an informal source. Informal notification seems to be more common at night than during the day and evening when the media play a more significant role.

The historical data do not reflect what is theoretically possible to achieve with specialized warning technology. The Lachman et al. study[46] of the Hilo tsunami indicated that within minutes 95% of the population heard the sirens that sounded the warning. Studies of nuclear power plants suggest that the portion of the population that hears warnings from the test soundings ranges from 60 to 95%, depending on the weather, season, and time of day.[47,48] Federal Emergency Management Agency (FEMA) tests of siren systems around nuclear power plants indicate a similar range, with a mean of 85% alerted by test soundings of sirens.[49] The larger problem is response to an alert mechanism. It is not known how many people actively seek information when they hear a siren. A recent study suggests that as many as 80% did not hear or understand the meaning of a siren during a recent chemical emergency.[50] The time that it takes to receive a warning message following the alert is largely unknown, but it is likely a logistic function that falls between the function for the hearing of a siren and that for systems based on official notification.

Other specialized warning systems capable of rapidly warning a high percentage of the population include tone-alert radios and automatic telephone dialers with steep penetration curves. Both systems can provide an alert and an instructional message. Recent experiences with tone-alert radios suggest that in approximately 70% of the households served by them, they produced a warning that was heard by people at home at the time.[49] The major problem with tone-alert systems is having an operable receiver. No reliable data exist on the actual performance of a telephone-based or aircraft system in a test or in an actual emergency.

MONITORING RESPONSE

The chief reason for monitoring how people are responding to a warning is to determine whether the warning system is guiding behavior in a manner consistent with the potential hazard and disaster risks. If people are engaging in actions that place them at greater risk, it may be the result of a poor warning. If the warning is not effective, adjustments in the warning process may be needed. These adjustments may include changing the content of the message, the frequency of dissemination, the channel of dissemination, the source of the information, or other basic facets of the warning process discussed in this article.

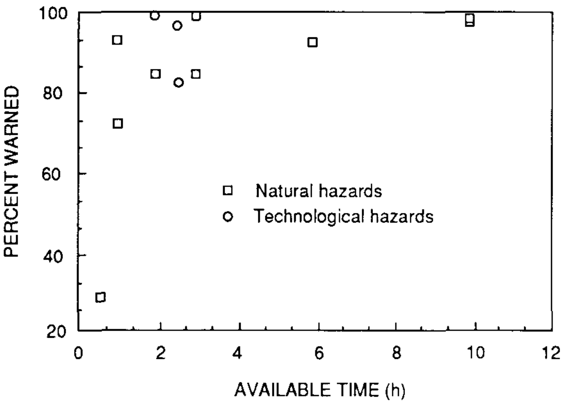

Fig. 2 Percent of public warned as a function of available warning time.

Methods of Monitoring Response

Several alternatives exist to monitor public response to disaster warnings. No one means is necessarily better than the next. At the same time, a mix of methods could be used in a particular event. A brief description of the methods follows.

Communication Lines to the Field. One way to gain feedback about response is to communicate with emergency workers, such as law enforcement officers in and on the periphery of the targeted warning area. This type of communication will allow for qualitative assessments of the warning response. If the advice is to seek shelter and people are observed on the streets, the clear signal is that not everyone is following the advice of the warning. If an evacuation is ordered, emergency workers can make spot checks to determine whether people are complying. One role of the Emergency Operations Center (EOC) is to organize these observations into a global picture to determine which actions are needed, if any, to communicate better warnings. In most disaster settings, these types of reports are made on an ad hoc basis. Some situations, however, may warrant more carefully planned feedback. In such cases it may be desirable to establish reporting requirements for some field personnel or a set of questions to ask while communicating with field personnel.

Systematic Observation. In some situations it may be desirable to have personnel assigned to observe systematically and in some cases quantitatively measure human response. This task can be completed in several ways. For a large-scale evacuation, it may be desirable to have traffic guides estimate the number of vehicles traveling on central routes. It also may be feasible to have shelter workers regularly report the number of people arriving at shelters. Other plans can be tailored to the specific risk situation.

Unobtrusive Measures. Unobtrusive indicators of public warning response may also be feasible. One obvious indicator is real-time traffic counters that measure vehicle flows from an area. These can be used to measure evacuation from risk areas, provided that the monitors are in the correct locations. Other possibilities include monitoring utility use rates, such as water consumption or electricity consumption. The latter approach, however, is largely hypothetical and has not been tested.

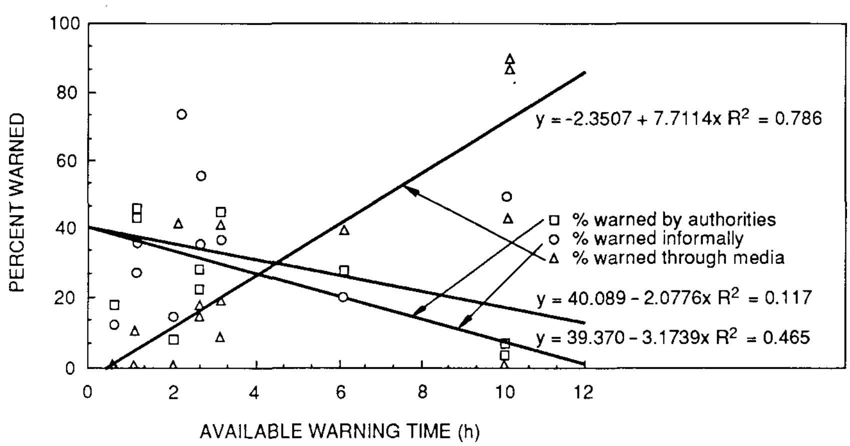

Fig. 3 Regression of percent warned by source as a function of available warning time.

Establishing a Monitoring System. A public monitoring system is an important part of a comprehensive warning plan, even though it may seem irrelevant before a disaster. Enough cases exist in which post-disaster audits show that if officials knew what was happening, a revised message or different warning strategy could have produced a more effective response, or, in some cases, saved lives. The concept of a monitoring system, however, has not been adopted in many emergency plans. It is an informal activity that takes place in some emergencies but is rarely labeled or formalized.

The first step to take in establishing public warning response monitoring capabilities is to review how information will feed into the EOC during an emergency and to assess whether it is adequate. If it is, then structure the nature of reporting to be done and determine by whom. Finally, make sure a backup means of communication exists. If the existing communication is inadequate, the addition of personnel to provide field reports may be necessary.

Potential problem areas, such as a narrow bridge on a major evacuation route, major freeways in an urban area, shelters in densely populated neighborhoods, or institutional facilities, may warrant a designated feedback mechanism.

CONCLUSIONS

In this article we have attempted to describe current thinking about some neglected aspects of building an effective warning system for radiological emergencies. The nuclear industry has, in complying with alert/notification regulations, the most advanced warning technologies in the country. Our key message is not to view warning systems as a set of technologies or equipment. Rather, warning systems consist of organizations, people, the communication process, the message, and the public, as well as the technology. Sophisticated warning systems could be of little use if they do not convey an effective message. Furthermore, in an actual emergency, a warning system is not unidirectional. It is important that responses to warnings be monitored to determine whether adjustments to the message are needed. The suggestions and evaluations provided in this review will help to improve the effectiveness of the warning process if they are put to use. Some of these suggestions have already been incorporated into the planning at several nuclear power plants, but at most plants they have not. They do not guarantee an effective system, but a fail-safe system is probably not achievable.

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Waste and Spent Fuel Management

Edited by W. H. Pechin

Activities Related to Waste Management

Compiled by E. G. Silver

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 the first quarter of 1989.

ACNW COMMENTS ON SEVERAL MATTERS

The Advisory Committee on Nuclear Waste (ACNW) has sent four letter reports to the Nuclear Regulatory Commission (NRC). Each will be briefly discussed and excerpted below.

Proposed Revision of 10 CFR 20 "Standards for Protection Against Radiation"

The ACNW undertook, at the request of NRC Commissioner Roberts, to study Section 20.205 of the proposed revision of 10 CFR 20 "Standards for Protection Against Radiation"; Section 20.205 deals with procedures for the control of long-lived radionuclides, typically those handled at fuel cycle facilities.[1] The staff proposed that, in view of other changes made, Section 20.205 be deleted. The ACNW comments on this proposal read, in part:

> As you know, the proposed rule published in the *Federal Register* on January 9, 1986 contained a new Section 20.205 which addressed the procedures noted above. The proposed section recommended a modified procedure that had been drafted in recognition of the difficulties in measuring (in a practical manner and with the required accuracy) air concentrations in restricted areas and the amounts of radionuclides in bioassay samples taken from workers whose intakes had been held at or below the permissible annual limits of intake (ALI). Although the proposed revision would have required licensees to design facilities so that air concentrations averaged over the year in restricted areas would be below the derived air concentration limits and would also have required that such facilities be operated in a manner that would ensure that any individual would be unlikely to have an intake from occupational exposure in any one year in excess of the ALI value, the modified procedure would have allowed licensees to permit doses to workers in excess of the limits in Section 20.201 as long as the sum of the internal and external effective dose equivalent would not have exceeded 5 rem, and the annual effective dose equivalent from certain specified internally deposited long-lived radionuclides would not have exceeded 3 rem.
>
> We believe that such a modified procedure is unacceptable. First, it would not be in accord with what we understand are the recommendations of either the International Commission on Radiological Protection (ICRP Publication 26, 1977) or the National Council on Radiation Protection and Measurements (NCRP Report No. 91, 1987). In addition, it is our interpretation that such a position would not be in conformance with the requirements outlined in the "Radiation Protection Guidance to Federal Agencies for Occupational Exposure," approved by President Reagan on January 20, 1987.
>
> Based on our review of this issue, we recommend that annual doses arising from the intake of long-lived radionuclides be limited to a dose commitment no higher than the annual dose limit of proposed Section 20.201. To make an exception for any specific group of radionuclides or licensees would, in our opinion, be

inappropriate. Hence, we concur with the NRC staff's recommendation to delete Section 20.205.

In addition, we recommend that the NRC encourage licensees to follow the guidelines contained in the Radiation Protection Guidance to Federal Agencies referred to above; namely, that record keeping include data on both the annual and committed effective dose equivalent, as well as on the cumulative (lifetime) dose.

Proposed Policy on Radiation Impacts "Below Regulatory Concern"

The NRC has issued an "Advance Notice of the Development of a Commission Policy on Exemptions from Regulatory Control for Practices Whose Public Health and Safety Impacts are Below Regulatory Concern." This advance notice asked for comments on several questions; the ACNW proceeded to consider these and other aspects of the proposed policy and furnished a letter response to the Commission in which each of the questions raised is cited and a response given, followed by some more general remarks. The letter reads, in part:[2]

The purpose of this report is to provide you with our responses to the several questions on which the proposed Policy Statement requested comments and to offer our comments on selected positions and/or premises outlined in the Policy Statement.

1. Justification of Practices

   In establishing its exemption policy, should the Commission exclude certain practices for which there appears to be no reasonable justification? In considering proposals for exemptions, should the Commission evaluate the social acceptability of practices?

   Response

   The ACNW believes that practices for which there appears to be no reasonable justification, particularly those that are considered to be of a "frivolous" nature, should be excluded from exemption. We concur with the staff in the examples that they cited for this category. At the same time, however, we would urge that the Commission recognize that what may be considered to be unjustified by one group may not be similarly regarded by others. We continue to believe that the Commission should exercise considerable care in reaching judgments on this matter.

2. Dose Limits and Criteria

   The Commission specifically seeks comment on the need for establishing a collective dose limit in addition to an individual dose criterion. If such a collective dose criterion is needed, what is the basis for this need? If the Commission decides that a collective dose criterion is needed, what approaches allowing truncation of individual dose in calculation of collective dose or weighting factors for components of collective dose would be appropriate? What alternatives should be considered for assessing societal impact?

   Response

   a. Collective Dose Criterion

      We continue to believe that a collective dose exemption level (or criterion) is necessary, but we also recognize that some flexibility should be allowed in setting that criterion. It is important to recall that annual doses to individual members of the public arising from an exempted practice will be estimated by use of models and assumed scenarios. These models will not be, and probably cannot be, validated. As a result, dose estimates derived through the application of such models will contain potentially important uncertainties. Further, exemption from controls also increases the range of possible exposure scenarios that can take place. This will add to the uncertain nature of the calculations. Although we are aware that estimates of collective population doses and determination of compliance are plagued by the same kinds of uncertainties, the additional constraints imposed by collective dose exemption levels should provide some further assurance of the continued acceptability of a practice that has been exempted.

      We believe that the magnitude of the collective dose criterion should depend on the associated dose rate to individual members of the public. As one possible approach, the Commission might consider that, for sources, practices, and/or devices that result in a dose rate as high as 10 mrem per year to individual members of the public, the collective dose criterion should be no greater than several hundred person-rem per year. For activities that result in dose rates well below 1 mrem per year, a collective dose criterion of several thousand person-rem per year might be considered.

   b. Truncation of Collective Dose

      Although a number of groups (such as the National Council on Radiation Protection and Measurements) have proposed individual dose rates (for example, 1 mrem per year or less) at which collective dose calculations should be truncated, we believe that such an approach would be strongly opposed by many groups within the public. We recommend that those responsible for calculating the impacts associated with a given practice being considered for exemption be required not only to provide an estimate of the total collective dose but also to provide data on the number of people within each dose rate range. Following this practice, all interested parties would be provided with detailed information on the contribution to the total collective dose by population groups in all dose rate ranges, including those in the extremely low ranges, and the Commission could take this information into consideration in deciding whether to exempt the practice. We believe the collective dose exemption approach suggested above will be helpful in making such judgments.

c. Alternatives for Assessing Societal Impacts

The Committee is not able to comment on the issues surrounding the social acceptability of a practice under consideration for exemption. We urge the Commission to proceed into this area with caution owing to the extensive and potentially unproductive polemics that could easily be generated.

3. Role of the As Low As Reasonably Achievable (ALARA) Criterion

In the Advance Notice of the Commission Policy, the NRC staff stated that, "If the dose is less than the below regulatory concern criteria, then the risk from a practice would be considered to be ALARA without further analysis."

Response

We believe that this statement is confusing and that it does not represent the approach that the NRC staff has indicated that it intends to follow.

In all cases, the staff has indicated that no practice would be exempted without a careful review of all details of its proposed application, that all practices will have to be justified, and that the proposed licensee will have to demonstrate that the given practice incorporates good radiation protection principles. For those practices that are exempted, there will be periodic, subsequent reviews to assure that they are properly implemented and that they do not result in dose rates to individual members of the public in excess of what was predicted.

Rather than characterize the exempted practice in terms of the ALARA criterion, we believe it would be better simply to say that the practice satisfies NRC radiation protection criteria, and its impacts have been found to be so small that the Commission has deemed it acceptable for the practice to be used or for the device or source to be released to the general public.

4. Designation of Exemption Levels

In discussions on this aspect of the Policy Statement, questions have been raised on several occasions on the individual dose rates that would be considered to be acceptable for exempted practices, sources, and devices. Although the Commission did not explicitly request comments on this matter, the Committee desires to offer the following remarks.

Response

First, it is important to note that there are practices, sources, and/or devices that result in exposure to the public for which exemptions have already been granted. These include consumer products, such as luminous dial watches exempted by the U.S. Nuclear Regulatory Commission, as well as items such as television sets that have been exempted by the U.S. Department of Health and Human Services. In addition, exposures resulting from the transportation of radioactive materials have been exempted through regulations of the U.S. Department of Transportation. In fact, according to studies of the National Council on Radiation Protection and Measurements (NCRP Report No. 95, December 1987), the average dose rate to individual members of the U.S. public arising from the use of consumer products (involving both radioactive materials and radiation generating machines) is currently at a level of 10 mrem per year. In short, this is not a new field.

Second, although the Policy Statement implies that some practices that could result in dose rates of as much as 100 mrem per year might be considered for exemption, we believe it is important to note that 100 mrem per year is the long-term dose limit for members of the public as recommended by the National Council on Radiation Protection and Measurements and the International Commission on Radiological Protection. It is also the limit recommended for members of the public in the revision being proposed by the NRC to Title 10, Part 20, of the Code of Federal Regulations, "Standards for Protection Against Radiation." A dose rate for individual members of the public approaching 100 mrem per year should not be viewed as an exemption level; rather, sources and practices that have the potential for causing dose rates in this range would have to be regulated. We foresee no conditions under which such sources, practices, or devices can be considered for exemption.

In terms of the exemption of practices, sources, and/or devices, it is our opinion that the limiting dose rate for individual members of the public as a result of exposures from all such exemptions should not exceed a value in the range of a few tens of mrem per year. Following this approach, and assuming that each person has the potentiality of being exposed to more than one such practice or source, then the exemption level per practice should be in the range of, at most, 1 to 10 mrem per year. We note that, in developing an exemption policy, the Commission is deciding how much of the 100 mrem per year dose limit for members of the public should be allocated to exempted practices, sources, and/or devices.

Since other government agencies have similar responsibilities, all such efforts should be well coordinated, and the total dose rate from all exempted practices must be well below (only a small fraction of) the dose limit.

5. Exposures to Multiple Practices

   The Commission seeks comment on whether individuals may experience radiation exposure approaching the limiting values through the cumulative effects of more than one practice, even though the exposures from each practice are only small fractions of the limit.

   Response

   The recommended dose rate exemption level of a few mrem per year for individual members of the public (arising from a single source, practice, and/or device) should provide reasonable protection against the inadvertent accumulation of annual doses in excess of the exemption level for individuals due to exposures to several exempted practices. Nevertheless, the Commission will need, in the long run, to guard against concentrations of exempted practices in localities and should include in its rules provisions that allow it to use judgment in this matter.

6. General Comments

   In addition to the comments above, the ACNW offers the following general comments.

   One requirement that the Commission should consider for inclusion in the exemption regulations is that for a source, practice, and/or device to be eligible for consideration, it must be "inherently" safe. That is to say, no accident scenario can be reasonably postulated that would result in doses to individual members of the public greater than a few mrem.

   The Commission should also emphasize that, even after the application of a practice has been justified and approval has been granted for its application and/or use, the situation will be reviewed periodically to ensure that the original conditions are being met and that the given practice, source, and/or device is still acceptable for exemption. This is currently a part of the Policy Statement. It should be emphasized.

   Equally important to the development of an exemption policy is the establishment of accepted exposure pathway scenarios, both for routine use of and accidents involving the practices, sources, and/or devices under consideration. This will require the development of environmental transport models and the derivation of secondary or derived guides (for example, concentration limits for specific radionuclides in low-level radioactive wastes that should be considered eligible for exemption), as well as the development of laboratory and/or field procedures for making the measurements necessary to confirm that the given practice, source, and/or device complies with the exemption levels.

   Finally, we believe that at this stage in the process one of the most important goals should be to develop a policy primarily designed for application on a case-by-case basis. It is also clear that procedural flexibility should be explicitly maintained. A Policy Statement incorporating both of these attributes can then guide the practices and, as experience is gained, both can be modified, if necessary, to lead to a more workable approach.

Activities of the Division of High-Level Waste Management

In late January 1989 the ACNW reviewed the activities of the NRC Division of High-Level Waste Management (DHLWM), especially with regard to the proposed High-Level Waste (HLW) repository at Yucca Mountain, and the role assigned to the ACNW in this effort.[3] The salient part of their letter of comments states:

> We found the discussions beneficial, and the NRC staff was fully responsive to our questions. We concluded that DHLWM has good leadership and their work is progressing well. We were particularly impressed by the efforts of the division director to keep the size of his staff modest and to monitor rather than duplicate the work of the U.S. Department of Energy (DOE).
>
> In terms of the work of this Committee concerning the NRC staff's ongoing review of the Site Characterization Plan (SCP) and their preparation of the Site Characterization Analysis for the HLW repository, we have concluded that our resources would best be directed to the activities noted below and intend to proceed in this direction:
>
> 1. An evaluation of the several "Review Plans" completed or being developed by the NRC staff to be used as guidance for its reviews, e.g., the Review Plans for the SCP and for Performance Assessment,

2. An evaluation of DOE's responses to the five "Objections" cited by the NRC staff concerning the Consultation Draft SCP; any additional areas of disagreement resulting from DOE's responses to the "Point Papers," which were prepared by the NRC staff; any substantive concerns raised by the state of Nevada; and any additional areas noted by the ACNW as being of special interest.

We also plan to review selected HLW rules, key NRC Technical Positions, and Regulatory Guides which are being developed within the NRC, as well as related plans and reports being developed by DOE. In addition, we plan to review relevant research under the direction of NRC, including the programs of the Center for Nuclear Waste Regulatory Analyses.

Comments on the West Valley Demonstration Project

The ACNW reviewed the West Valley Project with staff of the DOE and the New York State Energy Research and Development Authority to discuss, among other items, the procedures developed for solidifying decontaminated supernatant low-level wastes and testing the melter for vitrification of the HLW (Ref. 4). The West Valley Demonstration Project uses the contamination at this now-inactive fuel-reprocessing test facility to develop and test waste handling and management strategies. Their report states, in part:

> . . . the Committee concludes that the program is appropriately focused and that the results are favorable. Although there appears to be good communication between the DOE contractors and staff and the Nuclear Regulatory Commission (NRC) staff, there may be a need for additional input from the NRC staff in two areas:
>
> 1. Acceptance criteria for the vitrified high-level waste, including the enumeration of testing procedures to indicate conformance with these criteria, need to be identified by DOE for the waste producers, and these criteria, in turn, need to be reviewed by the NRC to determine if they are acceptable; and
> 2. Public health and safety criteria for the facilities and land areas being decontaminated and decommissioned as part of this project need to be established.
>
> We plan to schedule a visit to the West Valley site within the next six months.

At the end of 1988, DOE issued a report detailing the potential social and economic effects of siting the proposed underground HLW facility under Yucca Mountain in Nevada.[5] This report is responsive to a requirement of the Nuclear Waste Policy Act (NWPA) that the Secretary of Energy report to Congress on the potential impact of a repository on the local area and population. The NWPA lists 14 specific categories that DOE must evaluate, ranging from schools and police to tourism. DOE's conclusion was that at least 12 of the 14 categories would be affected if the Yucca Mountain HLW repository were built.

In the report, however, DOE differentiates between "effect" and "impact." A finding that a community may be affected means only that some change will occur as a result of the repository. An impact finding, however, indicates a change that will need to be mitigated through DOE or local action.

In formulating its report, DOE attempted to determine only whether or not the repository would have a measurable effect or impact in the study area. The report does not attempt to specify the timing or extent of potential changes. The Department said that this information will become available through ongoing monitoring of the waste disposal program.

The report considered the effect on four Nevada counties (Nye, Esmeralda, Lincoln, and Clark) as well as the effect on the state of Nevada as a whole. The Yucca Mountain site is located in the southern portion of Nye County, about 100 miles northwest of Las Vegas. On a statewide basis, DOE determined that the effects of the repository will be minimal. Several small communities may be impacted, according to DOE, but the Department believes that sufficient funds exist for mitigation. DOE's report stressed, however, that socioeconomic effects would vary with the capabilities and priorities of the local governments. Therefore DOE expects the initial evaluation of mitigation expenses to be made at the local level. Federal financial assistance may then be requested from DOE.

In the preparation of this report, DOE assumed that during the peak construction phase 2800 employees would be working at Yucca Mountain. This would decrease to 1965 employees during the repository's peak operating period. Only in Nye and Clark counties would there be sufficient numbers of new workers to impact public services. Within Nye and Clark counties, the report finds that the towns of Pahrump and Indian Springs will receive the largest population increases and that both may feel an impact in such areas as wastewater treatment, law enforcement, and fire protection.

In response to the possibility of accidents involving high-level radioactive wastes, DOE's report outlined the radiological emergency training courses developed in conjunction with the Federal Emergency Management Agency. According to DOE, seven courses of various orientation have been, and will be, held throughout Nevada to prepare local officials for the possibility of a radioactive accident. The report does not evaluate the specific impact of accidents related to the transportation of radioactive wastes because the routes such wastes would take have not been identified, DOE said. The report does cite, however, the need for additional radiation detectors statewide plus expanded radiation treatment facilities at local hospitals.

HOUSE BILL ON HAZARDOUS MATERIAL TRANSPORT INTRODUCED

In January 1989 Congressman H. B. Gonzalez (D-Tex.) introduced a bill, H. R. 506, to amend the 1974 Hazardous Materials Transportation Act (HMTA), which authorizes the Department of Transportation (DOT) to regulate the transportation of hazardous materials, including radioactive wastes and spent nuclear fuel.[6] The act allows DOT to preempt state and local regulations if they are inconsistent with federal regulations or unduly burden commerce.

Gonzalez said that the amount of hazardous materials being transported each year is increasing tremendously and that many localities are ill-equipped and lack sufficiently trained personnel to deal adequately with this growth. He added that existing rules and regulations concerning the transportation of these materials are at times unenforceable because of lack of personnel or resources and the difficulty in keeping state rules, regulations, and enforcement uniform.

The bill would amend Section 106(b) of the HMTA by requiring hazardous materials carriers to register with DOE every other year. In addition, the bill would not permit licensed carriers to ship hazardous materials through a city or town with a population of more than 50 000 without first notifying city officials of the shipment and receiving from them a transport route to follow.

The bill also calls for the imposition of a tax on the shipment of hazardous materials to be paid by the shipper registered with DOT. The tax, as defined in Section 103 of the Act, would be set at the rate of $0.025 per ton of hazardous materials.

The bill would also add a new section to the HMTA to establish a Hazardous Materials Trust Fund in the U.S. Treasury. Amounts equivalent to the taxes received by the Treasury after Sept. 30, 1990, will be appropriated to the fund.

Finally, Section 116 of the Act would allow the Secretary of DOT to make grants from the trust fund to cities to develop hazardous materials transportation plans and designated routes for shipment of such materials and to purchase equipment used in handling transportation accidents. The bill also requires the secretary to establish training centers for personnel responsible for responding to hazardous materials transportation accidents.

Several bills similar to H.R. 506 were introduced in the House last year, but questions concerning adequate funding for personnel training and the states' roles in adopting highway routes blocked passage of the bills.

H.R. 506 was jointly referred to the Committees on Energy and Commerce, Public Works and Transportation, and Ways and Means.

BOTH DOE AND NRC ANNOUNCE POLICIES ON PERFORMANCE MILESTONES

On Feb. 1, 1989, the NRC issued regulations that establish criteria and procedures for evaluating requests for emergency access to operating non-federal low-level radioactive waste disposal facilities.[7] Under the terms of the 1985 Low Level Radioactive Waste Policy Amendments Act (LLRWPAA)

> . . .individual states and regional compacts must take certain actions leading to the development of their own low-level radioactive waste disposal capacity within the periods of time specified in the Act. If these actions are not taken within the time frames specified, generators of low-level radioactive wastes within the non-complying state or regional compact may be denied access to existing disposal facilities after January 1, 1989.
>
> However, the Act authorizes the Commission to grant low-level waste generators or states emergency access to any of the operating non-federal low-level waste disposal facilities. In order to grant such a request, the Commission must find that such action ". . .is necessary to eliminate an immediate and seri-

ous threat to the public health and safety or the common defense and security . . ." and that ". . .the threat cannot be mitigated by any alternative consistent with the public health and safety, including storage of low-level radioactive waste at the site of generation or in a storage facility, obtaining access to a disposal facility by voluntary agreement, purchasing disposal capacity available for assignment or ceasing the activities that generate the low-level waste.

Under the new NRC regulation, a person seeking emergency access must submit detailed information to the Commission on the need for access to low-level waste disposal sites; the quantity, type and nature of the material requiring disposal; impacts on public health and safety or common defense and security if emergency access is not granted; the alternatives considered; and the process used to conclude that none of the alternatives are reasonable.

In making a determination that the circumstances described in a request for emergency access create a serious and immediate threat to the public health and safety, the Commission will consider:

(1) the nature and extent of the radiation hazard that will result from the denial of emergency access, including consideration of the NRC's standards for radiation protection contained in Part 20 of its regulations, any standards for the release of radioactive materials to the general environment that apply to the facility that generated the low-level waste, and any other Commission requirements that apply to the facility or activity for which emergency access is being requested; and

(2) the extent to which essential services such as medical, therapeutic, diagnostic or research activities will be disrupted by the denial of emergency access to waste disposal facilities.

In making a determination that the circumstances described create a serious and immediate threat to the common defense and security, the Commission will consider:

(1) whether the activity generating the wastes is necessary to the protection of the common defense and security (giving consideration to the views of the Department of Defense and the Department of Energy) and

(2) whether the lack of access to a disposal site will result in a significant disruption in that activity that will seriously threaten the common defense and security.

The new rule also sets out criteria for determining whether to grant "temporary" emergency access. The Act allows the Commission to authorize such temporary access for not more than 45 days, without considering available alternatives, if it concludes that the threat to the public health and safety or common defense and security warrants such action.

If the Commission determines that there is a need for emergency access, or "temporary" emergency access, it will then decide which operating non-federal low-level waste disposal facility should receive the waste, using criteria set out in the rule such as whether the waste and the disposal facility are compatible or whether a disposal facility has been previously designated to receive emergency access waste.

A proposed rule on this subject was published in the *Federal Register* on December 15, 1987. Changes made as a result of the comments received are mainly clarifying in nature. The procedures and the criteria to be used in making emergency access decisions are essentially unchanged.

In two separate announcements in the *Federal Register,* DOE and NRC explained how the two agencies view their policies regarding the performance milestones established under the LLRWPAA.

The LLRWPAA establishes milestones for the development of new disposal facilities in compact regions and states that do not currently have operating facilities. These goals must be met by the states or regions in order to receive rebates on a portion of the disposal surcharges paid by waste generators in their jurisdiction to the three operating disposal sites in South Carolina, Washington, and Nevada. Twenty-five percent of these surcharges are held in escrow by DOE until the milestones are met, at which time the funds will be returned to the states. The third milestone in the act, application for, or certification of, a site will occur on Jan. 1, 1990.

The January 23 DOE notice (54 FR 3106) states that compliance will be achieved when the Department receives from NRC a statement certifying that a completed license application for a low-level disposal site has been made or that the governor of a non-compact state has certified to NRC that a disposal site will be ready by Dec. 31, 1992. In addition, compliance may be achieved by a state through a bilateral agreement with another state in which a disposal facility already exists.

In the event a disposal application does not provide for the disposal of all low-level waste for which a state or compact is responsible, the statement should identify which wastes will be excluded and how they will be managed.

The DOE will return refunds to regional compacts rather than individual states if the license application accounts for the disposal of all waste within the compact's authority. Compacts may also receive refunds if the governors of all the states party to a compact submit letters of certification that a disposal site will be ready by 1993.

In its announcement, DOE stated that it will attempt to comply with the Act's stipulation that refunds be issued within 30 days of a state or compact's compliance. However, DOE noted that, because it does not control the administrative process by which it receives notification of compliance, the department cannot guarantee that the 30-day timetable will be met.

The notice by NRC was published on February 22 (54 FR 7616). It repeated the two potential methods of meeting the milestone but estimated that most states will file certifications rather than license applications.

The NRC also indicated that it will include any class A, B, or C waste that contains nonradioactive hazardous material in its definition of low-level waste and will expect the certifications to address disposal of this waste.

The technical content of the certifications should include: an estimate of the size and kinds of waste and who will generate it, a description of the proposed action for disposal of wastes, a statement that the proposed actions are consistent with NRC regulations, and the logistics of the proposed action.

MRSRC ISSUES STATUS REPORT TO CONGRESS

The Monitored Retrievable Storage Review Commission (MRSRC) issued a Feb. 22, 1989, status report to Congress on its activities since it was formed on June 14, 1988, pursuant to the 1987 NWPAA. Brief excerpts from the status report appear in the following text:

To date, the Commission has:

- Held briefing sessions on work done on the MRS prior to the creation of the Commission;
- Visited two sites for dry storage of spent nuclear fuel;
- Inspected work being done in four European countries on storage of spent fuel;
- Conducted five public hearings at which 82 persons testified and has received numerous written comments;
- Begun to outline the structure of the report;
- Formulated representative strategies which the Commission intends to evaluate;
- Developed criteria upon which the Commission's evaluation will be based; and
- Contracted for five technical studies to augment the staff's work.

Organization of the Commission

After taking office in June, 1988, the Commission organized administratively and began to assemble a staff. The Commission hired a small staff of professionals to perform research, supervise contracts, and assist the Commission in the preparation of the report. Day-to-day activities of the staff are managed by an Executive Director, who also serves as General Counsel.

Information Collection

In July 1988, interested parties, including DOE, NRC, members of Congress and Congressional staff, the General Accounting Office (GAO), the nuclear industry, the State of Tennessee, and environmental groups briefed the Commission on monitored retrievable storage work done before the Commission was created. . . .

Soon after its formation, the Commission determined it would be important to examine first-hand the work being done by utilities and others regarding the handling and storage of spent fuel. In early October, the Commissioners and the Executive Director visited Carolina Power and Light Company's H. B. Robinson Nuclear Project in Hartsville, South Carolina and Virginia Power Company's Surry Nuclear Power Station in Surry, Virginia. Although these are the only commercial nuclear power plants in the United States which have developed at-reactor dry storage facilities for spent fuel, other utilities are exploring the possibility of at-reactor dry storage. . . .

In addition to the U.S. site visits, the Commissioners and the Executive Director visited four countries in Europe—Sweden, the Federal Republic of Germany, France, and Switzerland—to learn about the European experience with spent fuel storage and to examine possible components of an interim storage system. . . .

Public Hearings

. . . To provide the opportunity for interested persons to present their views, the Commission held a series of public hearings in different sections of the country.

The Commission sent out more than 6,000 notices of the hearings to persons and organizations around the country, noticed the hearings in the *Federal Register*, and announced them over the wire services and in press releases sent to the regions in which they were being held. The hearings were well attended and have produced a wealth of information the Commission will consider during its deliberations. . . .

Sixty-three people testified as scheduled witnesses before the Commission during the five public hearings. Of the sixty-three, eighteen persons spoke in favor of an MRS, thirty-four spoke against, and eleven raised issues relevant to the need for an MRS but did not take a position on whether one should be built. . . .

With few exceptions, persons testifying before the Commission supported the siting and construction of a repository for the permanent disposal of high-level radioactive waste, whether or not an MRS is built. There were differences of opinion, however, as to the speed with which such an effort should be undertaken.

A majority of the representatives of the nuclear industry spoke in favor of an MRS, citing the need for such a facility so DOE could meet its statutory obligation to accept spent fuel by January 31, 1998. Industry advocates of an MRS argued that timeliness in the siting and construction of an MRS is critical, since the usefulness of an MRS facility would diminish if it were made operational on the same schedule as a permanent repository. . . .

[S]ome representatives of the nuclear industry stated that storage of high-level wastes could be accomplished at the reactor sites, and they favored that alternative. Generally, those expressing this opinion concluded that continued slippage in DOE's schedule for opening a repository made it highly unlikely that an MRS could be sited and constructed successfully within a reasonable time, especially in view of the "linkages" currently in the law between the repository and an MRS. These utilities asserted that it is more cost-effective and equally safe to store the wastes at reactor sites while the government moves as fast as possible to construct a permanent repository. . . .

Most of the environmental and citizen groups who testified were opposed to an MRS, preferring the alternative of continued at-reactor storage until a permanent repository is available to accept spent fuel. Spokespersons for environmental organizations expressed a wide range of reasons for opposing an MRS, including health, safety, transportation, cost, socioeconomic, lifestyle, ethical and moral factors. Almost all environmentalists expressed the concern that if an MRS facility were constructed, it would become a de facto repository. . . .

Most state and local government representatives refrained from taking a position on the MRS but raised concerns about 1) the transportation of high-level radioactive wastes through their regions, and 2) the emergency response capabilities of local communities along the transportation routes. They stressed the need for an early decision on whether to include an MRS in the nuclear waste management system, to allow sufficient time for planning and emergency response training along the proposed routes.

Private citizens spoke both for and against an MRS. The testimony of several citizens centered on the location of an MRS, even though the MRS Review Commission has no responsibility for recommending a site. . . .

. . . A few private citizens or citizen groups came before the Commission to promote the location of an MRS facility in their localities. They cited the need for economic growth in their communities, and said an MRS would not endanger a host community's overall health and safety.

Representative Strategies

The Commission has begun to outline the scope of its report, to identify the issues it intends to address, and to develop the criteria upon which its evaluation will be based. . . .

During its preliminary discussions, the Commission has identified four generic strategies representing many possible configurations of waste management systems which it is likely to evaluate in the final report. These strategies are sufficiently representative of the full spectrum of available options to enable the Commission to make a meaningful recommendation, but are concise enough to be susceptible to evaluation within the limited time and resources available to the Commission.

1. At-reactor storage until the repository is ready to accept spent fuel (No MRS);
2. Hybrid systems with a mix of at-reactor storage and MRS storage at regional MRS facilities until the repository is ready to accept spent fuel (Mix of at-reactor and one or more MRSs);
3. Storage at a central facility until the repository is ready to accept spent fuel (MRS-storage only); and
4. Processing and storage at a central facility until the repository is ready to accept spent fuel (Multi-function MRS).

The Commission intends to evaluate these strategies under a variety of scenarios regarding the integrated waste management system. . .The strategies will be evaluated with regard to these factors:

1. Overall safety and environmental impacts;
2. Effect on safe, efficient preparation of spent fuel for safe, permanent disposal (impact on repository design and construction; waste preparation; waste package design, fabrication, and standardization);
3. Transportation impacts (will include consideration of dual purpose and universal casks);
4. Flexibility and reliability of the national nuclear waste management program;
5. Economic efficiency;
6. Effects on public confidence in the national nuclear waste program;
7. Likelihood of meeting applicable regulatory requirements;
8. Likelihood of adverse impacts on reactor operations;
9. Equity of the system (e.g. regarding distribution of costs and benefits); and
10. Likelihood that DOE will be able to meet its contractual obligations.

DOE SUBMITS DRY CASK STORAGE REPORT TO CONGRESS

In March 1989 DOE's Office of Civilian Waste Management submitted to Congress a 130-page final report on at-reactor dry-cask storage of spent nuclear fuel.[8] The DOE finds such storage to be technologically feasible, safe, and environmentally acceptable; this provides another option for interim storage of spent fuel until a geologic permanent repository is opened sometime early in the next century.

The study also finds that DOE is not authorized to use the Nuclear Waste Fund to provide direct financial support to utilities for storage at reactor sites. However, the report says, "DOE will consider mechanisms whereby utilities can realize benefits resulting from at-reactor spent fuel management activities if such activities can be. . . beneficial to the overall waste management system."

Most spent fuel is currently stored under water in pools at reactor sites, but some utilities are facing the need for additional storage capacity. For this reason, utilities are considering several dry storage options for expanding their onsite storage capacity.

Utilities are also considering other technologies to expand their existing pool capacities through re-racking and fuel rod consolidation. For completeness, the study also evaluates these alternative technologies.

In conducting the study, DOE was required to consider such factors as costs, effects on human health and the environment, effects on the costs and risks of transporting spent fuel to a federal facility, and the extent to which the Nuclear Waste Fund can and should be used to provide funds for at-reactor storage.

NRC UPDATED ON WEST VALLEY PROJECT

On Mar. 29, 1989, DOE staff and representatives of the New York State Energy Research and Development Authority (NYSERDA) briefed the NRC Commissioners on the status of the West Valley Demonstration Project.[9]

In 1980 Congress passed the West Valley Demonstration Act authorizing DOE to carry out a nuclear waste management project to demonstrate that liquid waste from reprocessing spent nuclear fuel can be managed safely in the United States. DOE assumed control of the Western New York Nuclear Service Center, a closed commercial nuclear fuel-reprocessing facility, in West Valley, N.Y., in 1982 for this purpose. Upon completion of the project, the facility will be returned to the state energy authority.

At the West Valley site, DOE and the Westinghouse Electric Corp. have begun to convert liquid waste generated by the plant into cement and will soon begin to convert waste into glass. Officials from DOE say that they hope to demonstrate solidification and preparation of high-level waste for permanent disposal.

Although the Department is responsible for the construction and operation of the facility, NRC has the role of reviewing DOE's plans and consulting it on the following: plans for HLW removal, solidification, and preparation for disposal; plans for the decontamination of facilities used for HLW solidification; HLW waste form and containers to be used for disposal; plans for storage and disposal of low-level waste (LLW) and transuranic waste; and safety analysis reports and other information related to public health and safety. In addition, NRC is given access to the site to monitor DOE activities and is charged with prescribing the requirements for decontamination and decommissioning.

There are two phases to the project. Phase I, which was expected to be completed by the end of 1998, includes solidifying liquid HLW in a form suitable for transportation and disposal and developing containers suitable for permanent disposal. Phase II, which was expected to end in 2020, includes transporting solidified waste to a federal repository for permanent disposal, disposing of accrued low-level and transuranic waste, and decontaminating and decommissioning tanks, facilities, material, and hardware used in the project. Ten percent of the implementation costs are to be picked up by NYSERDA, the rest by DOE.

Since the processing began May 23, 1988, decontamination of factors between 5 000 and 150 000 have been achieved through zeolite beds, product acceptance has been rated at 99.93%, and the drum dose rates have been no more than 70 mrem/h at contact, compared to the 700 mrem/h design. As of March 10, 152 000 gal of liquid HLW had been processed and 2 914 cement drums produced.

According to current scheduling by DOE's West Valley Project Office, HLW processing will be done by the beginning of 1991 and civil structural construction by mid-1991. Sludge washing will immediately follow, as will electrical/mechanical construction. Sludge mobilization construction will begin in mid-1992, cold operation when all construction is completed in mid-1994, and finally hot operation in mid-1996.

For Phase II, HLW canister storage is expected at the beginning of 1999, as is the D&D program, and shipping is planned for 2010. Project site surveying and monitoring is set for 2016. "These are just base case figures," said J. E. Baublitz, Acting Director for DOE's Office of Remedial Action and Waste Technology. "We're looking to speed up the final stages of Phase II by retaining the glass logs in storage for a shorter time."

For decontamination, decommissioning, and closure of primary buildings, structures, and systems, DOE and NYSERDA are considering the following alternatives: (1) decontamination for unrestricted use; (2) decontamination and sealing for restricted access, surveillance, and site monitoring; (3) decontamination, demolition, and onsite disposal; (4) decontamination, demolition, and offsite disposal; and (5) no action, restricted access, surveillance, and site monitoring.

For decommissioning etc., of solids waste management or disposal units, the groups are reviewing (1) stabilization and closure; (2) exhumation, repackaging, and disposal; and (3) no action, restricted access, surveillance, and site monitoring.

The DOE and NYSERDA are also looking at proposed alternatives for the disposal of radioactive waste other than HLW: (1) onsite disposal; (2) offsite disposal; (3) interim storage pending availability of disposal capacity; and (4) no action, restricted access, surveillance, and site monitoring. And finally, the environmental impact statement (EIS) will include one of the following alternatives for transporting stored HLW for disposal: (1) early shipout to an interim storage site; (2) onsite storage awaiting availability of a licensed repository; and again (3) no action, restricted access, surveillance, and site monitoring.

Following DOE's presentation, T. K. DeBoer, Director of NYSERDA's Radioactive Waste Management Program, noted several concerns New York State has with the project. First, he said the state is "displeased" with the funds set aside in the fiscal year 1990 budget and added that this slippage in allocations can push back completion by 4 yr. Second, DeBoer explained that New York is wary of the availability of an HLW repository when the glass rods are ready for permanent disposal. And finally, DOE's criteria for HLW form acceptance specifications are "too restrictive," making them very difficult to meet, he continued. "Millions of dollars will have to be spent to meet these unreasonable and extremely wasteful specifications," DeBoer charged.

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Operating Experiences

Edited by G. A. Murphy

Steam Generator Tube Performance: Experience with Water-Cooled Nuclear Power Reactors During 1985

By O. S. Tatone[a] and R. L. Tapping[a]

Abstract: *The performance of steam generator tubes at water-cooled reactors during 1985 has been reviewed. In the survey, 73 of 168 reactors experienced tube degradation sufficient for the tubes to be plugged. The number of tubes plugged was 6837, or 0.28% of those in service. The leading cause of tube failure was stress corrosion cracking from the primary side. Stress corrosion cracking or intergranular attack from the secondary side and pitting were also major causes of tube failure. Unlike most previous years, fretting was a substantial problem at some reactors. Overall, corrosion continued to account for more than 80% of the defects.*

Steam generators are used in nuclear power plants to transfer heat created in the core to a secondary steam-raising circuit. They are large cylindrical tube-in-shell heat exchangers up to 20 m tall by 4 m in diameter. The several thousand thin-wall tubes carry the primary coolant, whereas the secondary coolant is on the shell side. Because of the severe operating conditions, the tubes are susceptible to a variety of failures. Consequently failures have been sufficiently numerous that significant costs have been incurred at some plants for mitigation and repair and for replacement power. Other plants have had reliable steam generator performance with good prospects for a 20- to 40-yr component lifetime.

Atomic Energy of Canada Limited has conducted detailed surveys of steam generator tube experience at water-cooled nuclear power plants outside the East Bloc countries since 1971 (Refs. 1 to 13). This report presents experience during 1985.

SUMMARY OF 1985 EXPERIENCE

During 1985, 168 reactors were surveyed, each with more than 100 effective full-power days (EFPD) of operation. These represent approximately 133 000 MW of net electrical generating capacity. They included 144 pressurized-water reactors, 23 pressurized heavy-water reactors, and 1 water-cooled, graphite-moderated reactor. The data were obtained by sending brief questionnaires directly to the utilities supplemented with information from other sources, including the trade and technical literature.

Table 1 lists the reactors at which steam generator tubes were plugged, the number of tubes plugged, the assigned causes of failure, and details of secondary water treatment and condenser performance. In this report tube defects or tube failures are defined to be any tubes plugged for whatever reason. This definition not only delineates the loss of heat transfer area but also helps to indicate trends and problem areas. Tubes may be plugged for any of several reasons, including actual primary-to-secondary leakage; indication of significant tube wall loss by nondestructive testing; or

[a]Chalk River Nuclear Laboratories.

Table 1 Experience During 1985[a]

Reactor	Tubes plugged	Assigned cause and location	Secondary chemistry control	Condenser cooling water	Condenser leaks	Comments
Asco 1	13	6 SCC(ID) in tubesheet; 7 others at tube supports	AVT	Fresh		4 tubes extracted
Beznau 2	8	5 SCC(ID) at U-bend; 3 fretting at U-bend	AVT	Fresh	2	39 tubes sleeved; 2 leakers
Biblis A	133	Wastage at U-bend	PO_4	Fresh		
Biblis B	22	Wastage at U-bend	PO_4	Fresh		
Blayais 1	2	1 SCC(ID) in tubesheet; 1 mechanical damage	AVT	Brackish		
Blayais 2	4	1 SCC(ID) in tubesheet; 3 mechanical damage	AVT	Brackish		
Blayais 3	1	SCC(ID) in tubesheet	AVT	Brackish		
Blayais 4	5	4 mechanical damage; 1 other	AVT	Brackish		
Bugey 2	98	94 SCC(ID) at U-bend; 4 mechanical damage	AVT	Fresh		
Bugey 3	1	SCC(ID) in tubesheet	AVT	Fresh		
Bugey 4	186	SCC(ID) at U-bend	AVT	Fresh		
Bugey 5	18	12 SCC(ID) in tubesheet; 6 SCC(ID) at U-bend	AVT	Fresh		
Calvert Cliffs 1	23	14 SCC/IGA at tubesheet; 2 fretting at U-bend; 7 others	AVT/CD	Brackish		
Calvert Cliffs 2	13	4 SCC/IGA at tubesheet; 2 fretting at U-bend; 7 others	AVT/CD	Brackish		
Cook 1	28	23 SCC/IGA at tube supports; 4 fretting at U-bend; 1 thinning at tube support (cold leg)	Boric acid	Fresh		
Cook 2	147	125 SCC/IGA at tubesheet; 16 SCC/IGA at tube supports; 1 thinning at tube supports (cold leg); 1 erosion; 4 others	Boric acid	Fresh		
Dampierre 1	191	90 SCC(ID) at U-bend; 84 SCC(ID) in tubesheet; 17 others	AVT	Fresh		
Dampierre 2	194	180 SCC(ID) at U-bend; 14 SCC(ID) in tubesheet	AVT	Fresh		
Dampierre 3	19	13 SCC(ID) in tubesheet; 2 mechanical damage; 4 others	AVT	Fresh		
Dampierre 4	181	178 SCC(ID) at U-bend; 3 others	AVT	Fresh		
Farley 1	1	Other	Boric acid	Fresh		
Farley 2	14	8 fretting at U-bend; 6 others at U-bend	AVT	Fresh		
Fessenheim 1	102	89 SCC(ID) at U-bend; 9 SCC(ID) in tubesheet; 1 fretting at U-bend; 3 others	AVT	Fresh		
Fort Calhoun 1	33	15 constriction at U-bend; 13 constriction at tube supports; 2 SCC/IGA at U-bend; 2 others; 1 undetermined	Boric acid	Fresh		
Genkai 1	142	SCC/IGA at tubesheet and tube supports	AVT/CD	Sea		147 tubes sleeved

(Table continues on the next page.)

Table 1 (Continued)

Reactor	Tubes plugged	Assigned cause and location	Secondary chemistry control	Condenser cooling water	Condenser leaks	Comments
Ginna	5	3 SCC/IGA in tubesheet; 1 thinning at tube support; 1 thinning at U-bend	AVT/CD	Fresh	2	69 tubes sleeved
Gravelines B1	193	179 SCC(ID) at U-bend; 12 SCC(ID) in tubesheet; 2 others	AVT	Sea		
Gravelines B2	188	179 SCC(ID) at U-bend; 6 SCC(ID) in tubesheet; 3 others	AVT	Sea		
Gravelines B3	97	90 SCC(ID) at U-bend; 4 SCC(ID) in tubesheet; 3 others	AVT	Sea		
Gravelines B4	169	SCC(ID) at U-bend	AVT	Sea		
Ikata 1	6	SCC(ID) in tubesheet	AVT/CD	Sea		
Indian Point 3	249	246 pitting at tubesheet; 2 thinning at tubesheet; 1 other at tubesheet	Boric acid	Brackish	8	635 tubes sleeved
Kewaunee	49	31 SCC/IGA in tubesheet; 14 SCC/IGA at tubesheet; 4 SCC/IGA at tube supports	AVT	Fresh		
KKS Stade	1	Wastage at tubesheet	PO_4	Fresh	1	
Ko-Ri 1	419	Pitting at tubesheet	AVT/CD	Sea		
Maine Yankee	36	20 pitting at tubesheet; 12 others at U-bend; 2 mechanical damage at tube support; 2 undetermined at tube support	Boric acid	Brackish	7	
McGuire 1	4	SCC(ID) in tubesheet	AVT/CD	Fresh	1	
McGuire 2	50	SCC(ID) in tubesheet	AVT	Fresh		2 leakers
Mihama 2	2	1 SCC(ID) at U-bend; 1 fretting at U-bend	AVT	Sea		1 tube extracted
Mihama 3	19	17 SCC(ID) in tubesheet; 2 fretting at U-bend	AVT/CD	Sea		
North Anna 1	86	77 SCC(ID) at tube supports; 9 SCC(ID) in tubesheet	Boric acid	Fresh	10	9 leakers
Oconee 1	3	Fatigue	AVT/CD	Fresh		1 leaker
Oconee 2	9	Erosion	AVT/CD	Fresh		
Oconee 3	18	15 erosion; 3 fatigue	AVT/CD	Fresh		2 leakers
Ohi 1	660	474 SCC/IGA at tube supports; 186 SCC(ID) in tubesheet	Boric acid	Sea		451 tubes sleeved, 1 tube extracted
Ohi 2	8	SCC(ID) in tubesheet	AVT/CD	Sea		
Palisades	8	Constriction at tube supports	AVT	Fresh	3	
Point Beach 2	54	37 wastage at tubesheet (cold leg); 11 SCC/IGA in tubesheet; 4 thinning at tube supports; 2 wastage at tube supports (cold leg)	AVT	Fresh		
Prairie Island 1	15	6 SCC/IGA in tubesheet; 5 fretting at U-bend; 4 thinning at tube supports	AVT	Fresh		1 tube extracted
Prairie Island 2	19	12 thinning at tube supports; 4 fretting at U-bend; 3 mechanical damage at tubesheet	AVT	Fresh		

Table 1 (Continued)

Reactor	Tubes plugged	Assigned cause and location	Secondary chemistry control	Condenser cooling water	Condenser leaks	Comments
Ringhals 2	146	144 SCC/IGA in tubesheet; 1 pitting at tubesheet; 1 fretting at U-bend	AVT	Sea		59 tubes sleeved, 5 leakers
Ringhals 3	8	2 fretting at preheater tube supports; 6 others	AVT	Sea	5	
Ringhals 4	2	SCC/IGA in tubesheet	AVT	Sea		
San Onofre 1	63	23 fretting at U-bend; 1 wastage at tubesheet; 39 undetermined at cold-leg tubesheet	PO_4	Sea		
San Onofre 2	330	249 fretting at U-bend; 2 mechanical damage at tubesheet; 79 others (manufacturing defects)	AVT	Sea	5	
San Onofre 3	258	234 fretting at U-bend; 24 others (manufacturing defects)	AVT	Sea	2	
St. Laurent B1	91	90 SCC(ID) at U-bend; 1 other	AVT	Fresh		
St. Laurent B2	182	180 SCC(ID) at U-bend; 2 SCC(ID) in tubesheet	AVT	Fresh		
St. Lucie 1	66	SCC/IGA	AVT	Sea		
St. Lucie 2	262	Fretting at U-bend	AVT	Sea		
Summer 1	280	163 SCC(ID) in tubesheet; 3 SCC(ID) at U-bend; 114 other SCC(ID)	AVT	Fresh		7500 tubes shot peened
Takahama 1	15	7 SCC/IGA at tube supports; 6 SCC/IGA in tubesheet; 2 SCC(ID) at U-bend	Boric acid	Sea		1 tube extracted
Takahama 2	456	SCC/IGA at tubesheet	Boric acid	Sea		9 leakers
Takahama 3	1	At tubesheet	AVT/CD	Sea		
Three Mile Island 1	335	247 SCC(ID) at free span; 69 SCC(ID) in tubesheet; 12 SCC(ID) at tubesheet; 1 SCC(ID) at tube supports; 6 other SCC(ID)	AVT/CD	Fresh		Caused by sulfate contamination
Tricastin 1	3	2 SCC(ID) in tubesheet; 1 other	AVT	Fresh		
Tricastin 2	88	85 SCC(ID) at U-bend; 1 fretting at U-bend; 2 others	AVT	Fresh		
Tricastin 3	89	SCC(ID) at U-bend	AVT	Fresh		
Tricastin 4	2	SCC(ID) in tubesheet	AVT	Fresh		
Trojan	28	SCC(ID) at U-bend	AVT/CD	Fresh	1	1 leaker
Yankee Rowe	108	45 wastage at tubesheet; 41 SCC/IGA at tubesheet; 22 others	AVT	Fresh		
Zion 1	74	64 SCC/IGA in tubesheet; 9 wastage at tubesheet; 1 other	AVT	Fresh		
Zion 2	4	2 fretting at U-bend; 1 SCC(ID) at U-bend; 1 wastage at tubesheet	AVT	Fresh		

[a]Abbreviations used:

AVT	All-volatile treatment
CD	Condensate demineralization
PO_4	Phosphate treatment
SCC(ID)	Primary-side stress corrosion cracking
SCC(OD)	Secondary-side stress corrosion cracking or intergranular attack (IGA)

because the tubes are in a region of the steam generator that experience has shown to be susceptible, and thus it is more economical to plug the tubes during a scheduled outage than to risk a forced outage. Note that the number of tubes having primary-to-secondary leakage is a small proportion of the total number of tubes plugged.

During 1985, 0.5% of tubes plugged had actually leaked. This figure is somewhat less than the experience reported in previous surveys. Also, 0.1% of the tube sites were plugged because they were extracted for destructive metallographic examination. Many of these were sound tubes. However, the majority of tubes were plugged because of indications of wall penetration by nondestructive analysis or because they were located in failure-prone regions in some steam generator designs. Also, in recent years effective methods have been devised to extend the heat-transfer life of defective tubes by installing sleeves of new material. Hence the number of tube failures is somewhat underestimated when measured by the number of tubes plugged.

Steam generator tubes were plugged at 73 reactors during 1985; this represents 43.5% of the reactors in the survey. The number of tubes plugged was 6837, or 0.28% of the tubes in service. These figures are comparable to the experience of previous years, both for the number of plants affected and the number of tubes plugged. The percentage of plants requiring tube plugging has always been between 31 and 47% of reactors in the survey. The percentage of tubes plugged has usually been less than 0.5% of those in service.

HISTORICAL PERSPECTIVE

The year-by-year history of tubes plugged since 1971 is given in Table 2. Twenty reactors were added to the survey in 1985, which is the largest single year increase since the survey began. The percentage of tubes plugged is up slightly compared with recent years, but the percentage of reactors with tubes removed from service is about the same as in previous years. Of the 6837 tubes plugged in 1985, 86% were from 27 of the 168 reactors surveyed. Since 1973, when 0.9% of in-service tubes were plugged (primarily because of phosphate wastage), the annual tube plugging incidence has remained from 0.1 to 0.4%. By the end of 1985, 44 897 tubes had been plugged, which is 1.8% of the 2.5 million in service. That percentage of total tubes plugged has been typical since about 1980.

Figure 1 summarizes the major causes assigned to steam generator tube defects since 1972. Phosphate wastage was the dominant reason for plugging from 1973 to 1975, followed by denting from 1976 to 1980. Pitting emerged as a new failure mechanism in 1981, along with an increase in the incidence of primary-side stress corrosion cracking [SCC(ID)]. The incidence of pitting has remained relatively stable at less than 10% since 1982, but the incidence of SCC(ID) has been on the increase since 1983, which accounts for nearly 45% of tube plugging in 1985. Secondary-side stress corrosion cracking [SCC(OD)] has been decreasing since 1983 as a cause of tube plugging. Note, however, that several steam generators that were replaced had a high incidence of SCC(OD). Also, sleeving has been used effectively to avert tube plugging because of this mechanism. Nevertheless, SCC(OD) still accounts for more than 24% of tubes plugged. Corrosion continues to account for more than 80% of all defects.

The SCC(OD) tends to occur in low-flow areas of steam generators and in locations where impurities can concentrate, such as under sludge deposits and in tube-to-tubesheet and tube-to-tube support plate crevices. Figure 2 indicates areas of a recirculating steam generator in which corrosion and fretting have occurred. Figure 3 illustrates trouble spots associated with once-through steam generators.

The SCC(ID) was initially associated with high stresses arising from denting. Recently, SCC(ID) has been associated with the U-bend area and, in particular, the innermost tubes (tightest radii). Of the 3063 tubes plugged in 1985 as a result of SCC(ID) indications, 1924, or 62.8%, were associated with U-bends and 613, or 20.0%, were in or near the tubesheet. Tight U-bend radii and the roll-transition region in the tubesheet are areas of high residual stress that are more susceptible to SCC(ID) than the remainder of the primary side of the steam generator. Many of the tubes that were plugged because of the occurrence of SCC(ID) at the U-bend were plugged as a precaution. The practice has been to plug either all first-row tubes or all first- and second-row tubes when SCC(ID) is detected in any of them. Hence the number actually displaying eddy-current indications is relatively small.

Table 2 Tubes Plugged by Year

	Reactors			Tubes		
Year	In survey	Tubes removed from service	Tubes removed from service, %	In survey	Plugged	Plugged, %
1971[a]	24	15	62.5	168 972	1 007	0.60
1972	32	11	34.4	321 380	881	0.27
1973	39	12	30.8	435 187	3 874	0.89
1974	51	23	45.1	601 047	2 002	0.33
1975	62	22	35.5	788 147	1 677	0.21
1976	68	23	33.8	864 261	3 757	0.43
1977	79	33	41.8	1 079 559	4 339	0.40
1978	86	32	37.2	1 195 057	1 267	0.11
1979	93	39	41.9	1 308 868	2 814	0.21
1980	96	40	41.7	1 358 712	1 902	0.14
1981	110	46	41.8	1 553 674	4 692	0.30
1982	116	54	46.5	1 642 535	3 222	0.20
1983	132	47	35.6	1 845 426	3 291	0.18
1984	148	63	42.6	2 081 313	3 335	0.16
1985	168	73	43.5	2 436 578	6 837	0.28

[a]Inclusive to 1971.

The performance of the steam generators to the end of 1985 is summarized in Fig. 4 as a log-log plot of failure incidence (cumulative tubes plugged per 100 tubes in service) vs. EFPD. (This figure is based on the data in Appendix B of Ref. 14.) Each point on the figure represents one reactor. Three diagonal lines representing failure rates (F) of 0.01 to 1% per effective full-power year (EFPY) have also been drawn. Points lying below $F = 0.01$ indicate units with tube failure rates of less than 0.01% per EFPY; this represents highly reliable steam generator performance. In contrast, units with $F > 1\%$ per EFPY (above the line labeled $F = 1$) may suffer forced denting or require large-scale sleeving or replacement of steam generators well before the design life of the reactor.

Seven reactors have, in fact, replaced their steam generators. Most had failure rates of 1 or greater per EFPY. Several other steam generator replacements are planned. Those with replacement steam generators are Obrigheim, Point Beach Nuclear Plant Unit 1, H. B. Robinson Plant Unit 2, Surry Power Station Units 1 and 2, and Turkey Point Plant Units 3 and 4.

Figure 4 illustrates the widely varying failure rates of steam generator tubes with similar service experience. In contrast to units that have replaced steam generators (all with EFPD of 1000 to 4000), several reactors had more than 2000 EFPD with either no tubes plugged or only 1 tube plugged. These were Bruce 1, KKU Unterweser, Loviisa 1, N Reactor (in the Alloy 600 tubes), Pickering 1, Pickering 3, and Pickering 4. The experience of these units, with a variety of materials and water chemistry, indicates that there is room for substantial improvement in nuclear steam generator technology, whether it be by design or improved operating techniques or both.

ASSIGNED CAUSES OF 1985 TUBE DEFECTS

The major causes assigned to tube defects in 1985 (Table 3) were SCC(ID) and SCC(OD)/intergranular attack (IGA). The number of reactors affected by SCC(ID) and SCC(OD) in 1985 was 55, up from 42 and 26 in 1984 and 1983, respectively.

Primary-Side Stress Corrosion Cracking

In 1982, half the failures assigned to SCC(ID) occurred at Three Mile Island Nuclear Station Unit 2 and were associated with sodium thiosul-

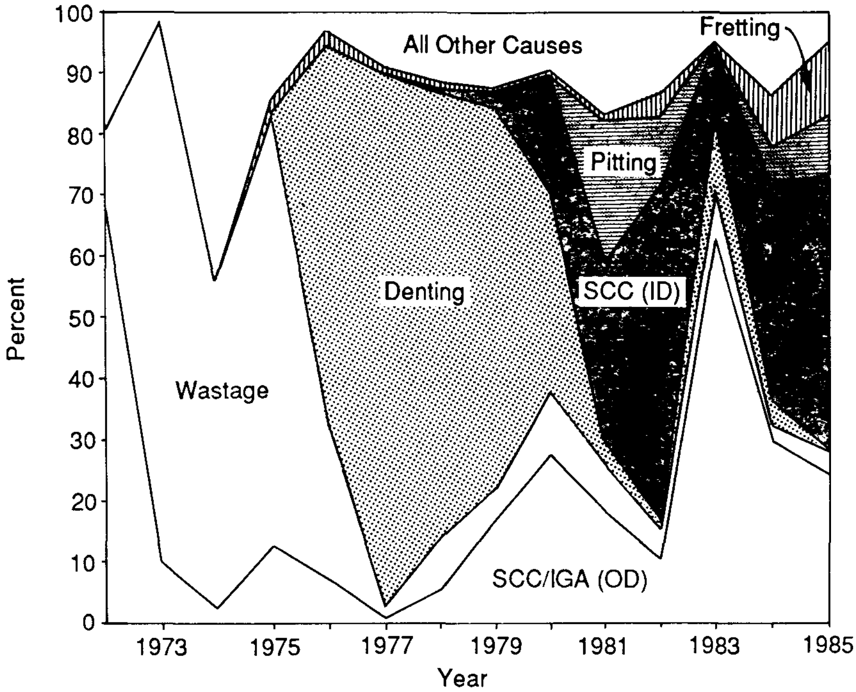

Fig. 1 History of tube failure mechanisms (note that more than 80% of the defects have been caused by corrosion). SCC, stress corrosion cracking; IGA, intergranular attack.

fate contamination of the reactor primary coolant.[15] In 1983 and 1984, the primary-side assigned failures were associated with SCC of highly stressed Alloy 600 in high-purity water—a phenomenon first reported by Coriou et al.[16] and subsequently confirmed by others.[17] Tight U-bends and the roll-transition region are areas particularly susceptible to this mechanism. In 1985, a further increase in defects assigned to this mechanism occurred, now accounting for 44.8% (3063 tubes) of assigned tube defects (vs. 11.7 and 35.4% in 1983 and 1984, respectively). Steam generators tubed with Alloy 800, stainless steel, and Monel 400 have not experienced this type of cracking after up to 14 yr of operation. Also, of the 22 reactors known to have thermally treated Alloy 600 tubes (e.g., at 705°C for 10 h), none has yet shown this type of failure mechanism.

Remedial and preventive measures for units tubed with mill-annealed Alloy 600 have focused on reducing residual stresses, with roto peening and shot peening becoming particularly common for the roll-transition region. Local and global heat treatments have also been considered, and sleeving is a practical technology for plants experiencing primary-side attack.[18]

Secondary-Side Stress Corrosion Cracking and Intergranular Attack

The SCC(OD) and IGA are grouped together here because they often occur in the same location and both are believed to be associated with an alkaline environment. In 1985, 18 reactors had plugged tubes as a result of SCC(OD)/IGA indications, compared with 13, 14, 14, and 14 in 1981 to 1984, respectively. The percentage of reported tube defects in 1985, 24.2%, was lower than that in 1984, continuing a trend to decreasing incidence of this type of attack since 1983. Note that since 1981, 5 of the 13 to 16 reactors reporting SCC(OD)/IGA have repeated in at least four of the five years. These reactors were Genkai 1, Robert Emmett Ginna Nuclear Power Plant, Ringhals 2, Takahama 2, and Point Beach 2. Most of the others have reported SCC(OD)/IGA in at least two of those five years. Of the 16 reporting in 1985, 9 were repeaters from 1984, which suggests that, once corrosive conditions capable of causing SCC/IGA develop on the secondary side of a steam generator, it is difficult to eliminate the problem.

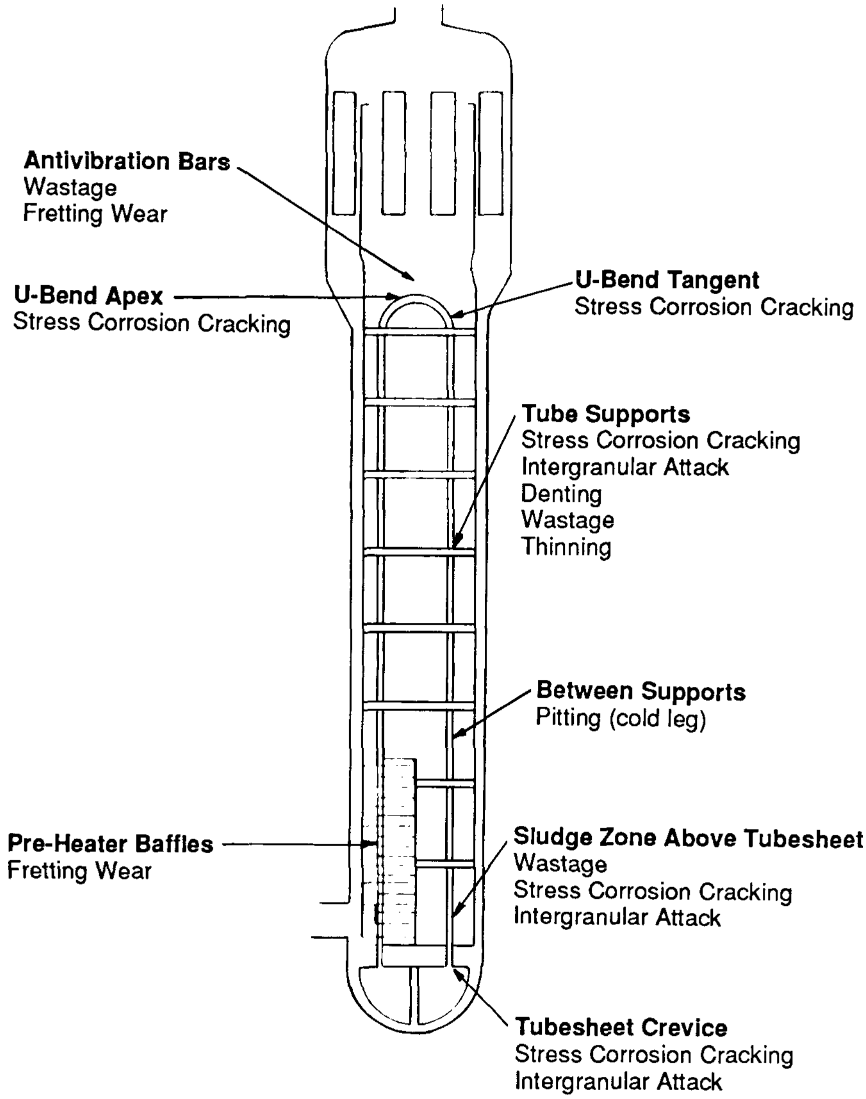

Fig. 2 Schematic diagram of a recirculating steam generator showing failure mechanisms and locations.

The SCC(OD)/IGA led to sleeving repair at seven reactors during 1985. Sleeving is a common method of repairing tubes affected by this type of corrosion, and because a sleeved tube is not considered a defect, in this review the data will underestimate the extent of SCC(OD)/IGA. SCC(OD)/IGA are frequently reported to occur in the crevices between tube and tubesheet or tube and tube support plate. The tubing is not usually sensitized, nor is prior phosphate treatment a prerequisite for this type of attack. Examinations of tubes removed from steam generators indicate that the aggressive environment is most likely alkaline, and, in the absence of prior phosphate treatment, the source of the alkalinity is most likely condenser cooling water. However, of the 16 stations that plugged tubes in 1985 as a result of SCC(OD)/IGA, several have not reported condenser leaks in the past 3 yr and have not used prior phosphate water chemistry control (Genkai 1, Ohi 1, Ringhals 4, and Calvert Cliffs Nuclear Power Plant Units 1 and 2). The others have all reported condenser leaks over the past 3 yr (6 stations) or used prior phosphate chemistry control (7 stations) or both (Ginna, Kewaunee Nuclear Power Plant, and Point Beach 2). It is not

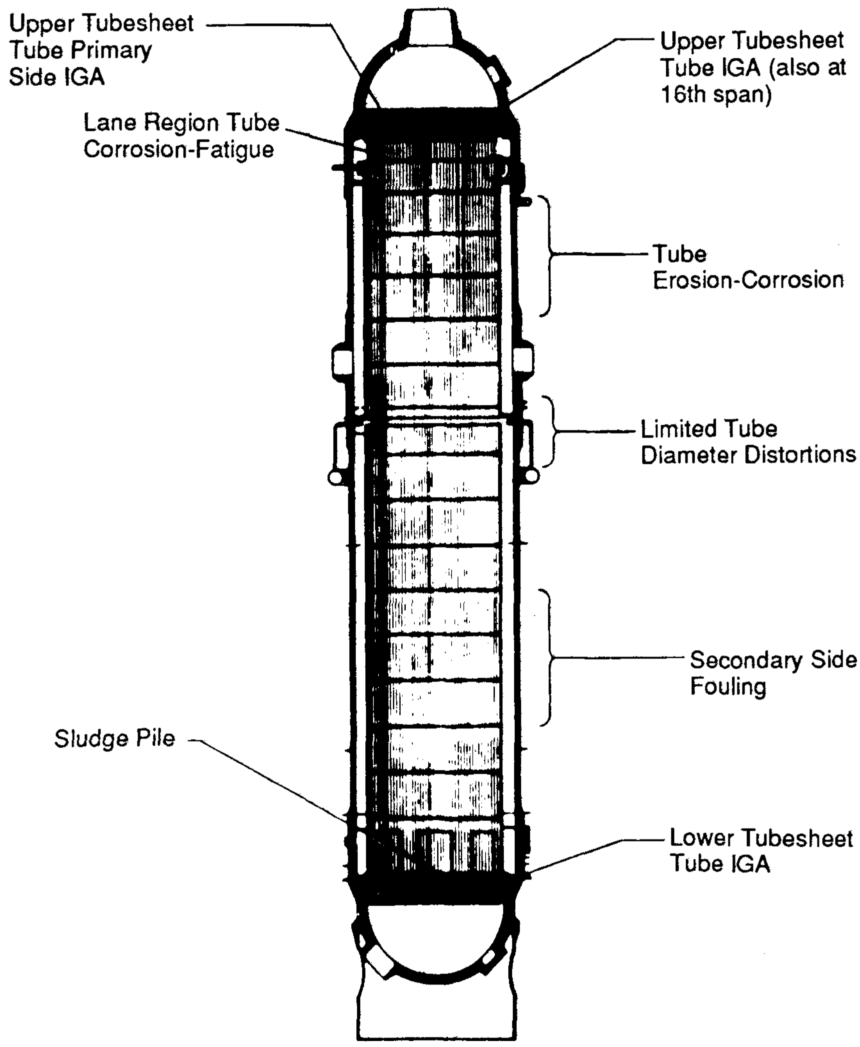

Fig. 3 Schematic of a once-through steam generator showing failure mechanisms and locations. IGA, intergranular attack.

known how long SCC(OD)/IGA takes to incubate following a condenser leak nor how long phosphate hideout remains a concern.

Pitting

Pitting generally occurs on the cold legs of steam generators in the presence of oxidizing ionic copper solutions. Cold-leg pitting was reported at four stations in 1985, compared with four, one, two, and five in 1981 to 1984, inclusive. Thus the incidence of pitting has remained low and relatively constant over the past 5 yr, although the number of tubes plugged at affected plants can be quite high.

Phosphate Wastage

Phosphate wastage was a reason for tube plugging in eight reactors in 1985, compared with eight, eight, four, and four from 1981 to 1985 inclusive. Thus the incidence of phosphate wastage is remaining relatively constant. Eleven reactors practiced phosphate chemistry control in 1985. The incidence of wastage in 1985 was new to only one station, Yankee Rowe Nuclear Power Station. Of the other five with plugged tubes because of phosphate wastage, two, Biblis A and KKS Stade, have had plugged tubes for this reason in every year since 1981. All six had been on phosphate chemis-

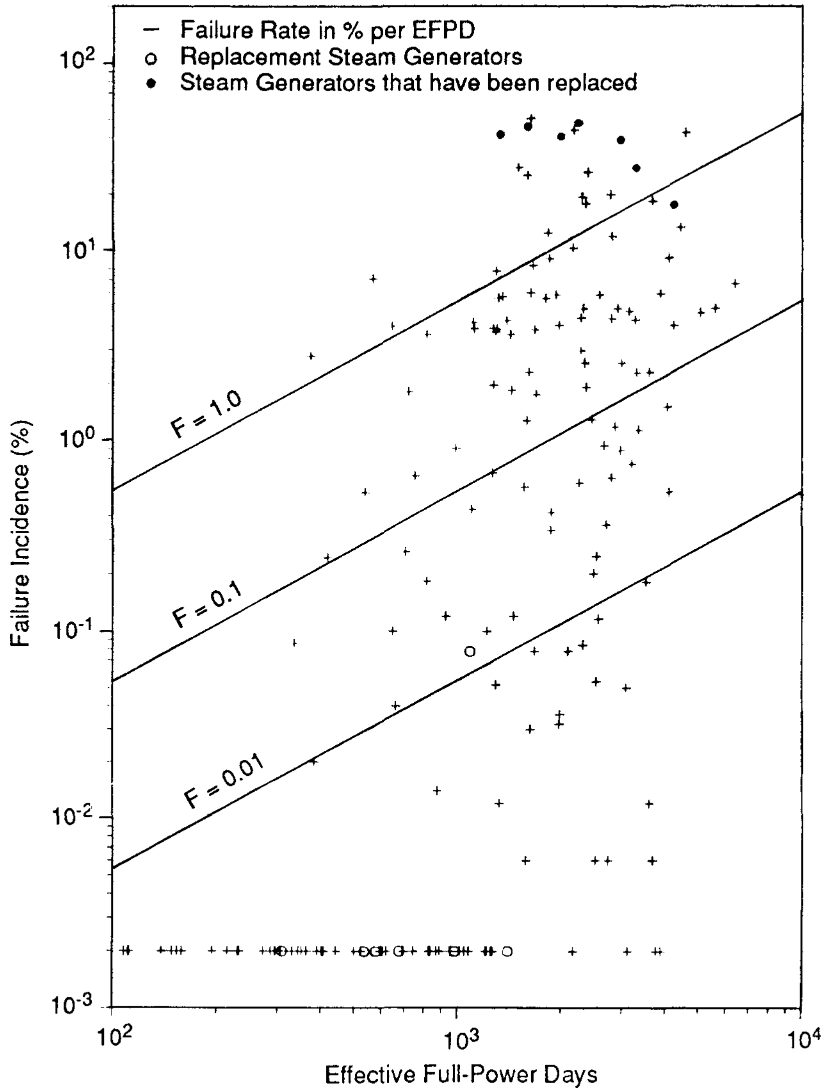

Fig. 4 Cumulative steam generator tube performance. Log-log plot of failure incidence vs. effective full-power days (EFPD) at the end of 1985. Reactors with F greater than 1 may require large-scale repairs or steam generator replacement.

try control at some point; four (Biblis A and B, KKS Stade, and San Onofre Nuclear Generating Station Unit 1) still practice phosphate chemistry control. Over the past 5 yr, 1981 to 1985 inclusive, phosphate wastage has been reported at 16 different stations. These 16 account for the total of 34 stations with plugged tubes as a result of phosphate wastage; this represents an average of about two incidents per plant over the 5-yr period.

Denting

Only two stations reported tube plugging because of denting in 1985, down from seven in 1984. Over the past 5 yr, the incidence of denting has remained consistently low (six, four, and two in 1981, 1982, and 1983, respectively), and denting now is a minor cause of tube defects. Only 36 tubes were plugged because of denting in 1985; this reflects a consistent decrease in tubes plugged since 1981. One of the stations that plugged because of denting in 1985 was Palisades Nuclear Plant, which has some drilled carbon-steel support plates and at one time operated on phosphate control. The occurrence of denting at Fort Calhoun Station Unit 1 in 1984 and 1985 is surprising because phosphate treatment was never used. The degree of deaeration may be important in this case.

Thinning

Thinning, although not a problem in many reactors, has shown a small but steady increase in incidence recently, with seven reactors reporting this type of corrosion in 1985, compared with four, two, three, and four from 1981 to 1984 inclusive. Thinning occurs in the cold legs of some steam generator tubes at the intersection with the support plate. Tubes subjected to destructive examination have shown areas of broad pitting or localized wastage, which are believed to be associated with ingress of resin fines in the steam generators, but the precise mechanism has not been identified.

Erosion

Tube plugging as a consequence of erosion took place in three stations in 1985. Two of these were Oconee Nuclear Station units, and several of these have a prior history of erosion–corrosion. In once-through steam generators, erosion is usually found in the area of the untubed lane and results from tube vibration and entrainment of magnetite particles. Only 25 tubes were plugged for this reason.

Fretting

Fretting incidence increased dramatically from a historical average of 6 or 7 reactors since 1981 (6, 7, 6, and 8 in 1981 to 1984 inclusive) to 18 reactors in 1985. Of these 18, 9 have had a prior incidence of fretting since 1981: in particular, Beznau 2, Prairie Island Nuclear Generating Plant Unit 2, and San Onofre 2. In 1984, San Onofre 2 plugged 249 tubes because of fretting at the diagonal U-bend supports. In 1985, San Onofre 2 plugged another 247 tubes for the same reason and was joined by St. Lucie Plant Unit 2 and San Onofre 3, where 262 and 234 tubes, respectively, were plugged. These three stations accounted for 743 of the 806 tubes plugged because of fretting. If San Onofre 1 is also included, the three San Onofre units accounted for 504 of the 806 tube pluggings.

Table 3 Assigned Causes of 1985 Tube Defects

Cause	Number of reactors affected	Number of tube defects	Tube defects, %
SCC(ID)[a]	37	3063	44.8
SCC(OD)/IGA[b]	18	1655	24.2
Pitting	4	686	10.0
Phosphate wastage	8	251	3.7
Denting	2	36	0.5
Thinning	7	26	0.4
Erosion	3	25	0.4
Fretting	18	806	11.8
Fatigue	2	6	0.1
Mechanical damage	8	21	0.3
Undetermined	3	42	0.6
Other	26	220	3.2

[a]Primary-side stress corrosion cracking.
[b]Secondary-side stress corrosion cracking or intergranular attack.

Fatigue

In 1985, two stations plugged six tubes because of fatigue, which is rougly the historical average of the past four or five years. Fatigue may be a generic, although very small, problem at Oconee units (along with erosion) and may also be related to the tube microstructure because tubing at both Bruce (fatigue in 1983) and Oconee units has a similar heat treatment (stress relief) that leaves tubes in a partially sensitized condition. Of course, design features conducive to fatigue may override microstructural details.

Mechanical Damage

Mechanical damage accounted for 21 tube pluggings at 8 stations in 1985. This level of incidence is somewhat lower than in past years. The cause of mechanical damage usually was not specified, but loose components/debris was the cause of damage in one unit.

Others

Note that in 1985, 103 tubes were plugged at San Onofre 2 and 3 because of manufacturing defects.

LOCATION OF DEFECT INDICATIONS REQUIRING TUBE PLUGGING IN 1985

Table 4 shows the assigned locations of tube defects requiring plugging in 1985. As in past years, corrosion indications on the secondary side occurred in areas of restricted flow, such as within the tubesheet crevices under sludge deposits, and in the crevices formed between tubes and tube support plates. The numbers in parentheses are the reactors and the tubes that had indications of SCC(ID). This problem invariably occurs at highly stressed regions, such as tight U-bends and roll-transition regions within or at the tubesheet. The

Table 4 Location of Pluggable Indications in 1985

Location	Reactors affected	Tubes plugged	Tubes plugged, %
Within tubesheet	33 (25)[a]	949 (682)	13.9
U-bend	45 (21)	2919 (1924)	42.7
Above tubesheet	23 (1)[b]	1494)[b]	21.9
Tube supports	20 (2)	661 (78)	9.7
Other	31 (2)	800 (367)	11.7
Undetermined	5	14	0.2

[a]Primary-side defects are given in parentheses [e.g., 33 (25) indicates that, of 33 reactors with indications at that location, 25 had primary-side indications].

[b]Caused in once-through steam generator by sulfate contamination of primary coolant.

total number of tubes plugged as a result of indications at or above the tubesheet does not include Genkai 1, where the 142 tubes were plugged because of SCC/IGA indications near the tubesheet or at tube support plates, but the number at each location was not given. Further, tubes plugged because of erosion, mechanical damage, and other mechanisms might also contribute to the failures listed as "above tubesheet" but have not been included there because of a lack of specific information. These tube pluggings are included in "other" or "undetermined."

SECONDARY WATER CHEMISTRY CONTROL

All-volatile treatment (AVT) with or without condensate demineralization (CD) is the secondary-water chemistry control choice of most of the stations reporting in 1985. Of the 168, 11 remained on phosphate and 8 were on boric acid chemistry control. All the stations on boric acid had switched from AVT, where one had switched from boric acid to AVT (Ko-Ri 1, 1982). Of the 149 stations indicating that they are on AVT, 127 have not used either phosphate or boric acid previously. The use of condensate demineralizers is generally associated with stations using brackish water or seawater for condenser cooling, but 17 of the 48 stations with CD use freshwater cooling. Seven of these have once-through steam generators.

Table 5 summarizes these data. Note that the stations using phosphate or boric acid remain a small but constant minority: in 1983 and 1984, there were 12 and 7 stations on phosphate and boric acid, respectively.

The relationship between secondary-water chemistry and secondary-side corrosion defects is illustrated in Table 6. Care must be taken in drawing conclusions from Table 6, particularly when considering data from stations that have changed their chemistry control. Restricting attention to the stations that have not changed their chemistry control suggests that AVT is better than AVT+CD, which, in turn, is better than phosphate. This suggestion does not take into account the differences in tubing or cooling water; for instance, most stations practicing AVT+CD use seawater cooling, and most of those on phosphate have Alloy 800 tubes.

Further analysis of the data is not warranted because of other systematic variations or insufficient data for a statistically meaningful interpretation. Similarly, an analysis of plugging rate as a function of a plant's ability to meet the Steam Generator Owner's Group guidelines[19] is beyond the scope of this review.

STEAM GENERATOR TUBE MATERIALS

Table 7 lists tubes plugged in 1985 as a function of tube material. Alloy 600 continues to exhibit a susceptibility to more types of degradation than any other material and consequently has the greatest plugging rate, 0.35%. This figure is higher than that in 1983 or 1984 (0.22 and 0.19%, respectively) but within the range 0.2 to 0.4% encountered over the past few years. Conversely, Monel 400 continued free of tube plugging and corrosion, as has been the case since these reactors went into service. Alloy 800 continues to show a low incidence of tube plugging, all of it (156 tubes in 1985) associated with wastage. All but one of

Table 5 1985 Secondary Water Chemistry vs. Type of Cooling Water

Cooling water	Number of reactors				
	AVT[a]	AVT + CD[b]	Phosphate	Boric acid	Total
Fresh	71	17	8	5	101
Brackish	4	12	0	3	19
Sea	26	19	3	0	48
Total	101	48	11	8	168

[a]AVT, all-volatile treatment.

[b]CD, condensate demineralization.

Table 6 Secondary Water Chemistry vs. Corrosion Defects[a] in 1985[b]

			Reactors			Tubes			
Chemistry in 1985	Prior chemistry	Year of changeover	In survey	With tubes plugged	With tubes plugged, %	In survey	Plugged	Plugged, %	Assigned failure mechanism
AVT	Phosphate	1974 to 1975, 1981	17	10	58.8	149 830	899	0.600	SCC/IGA,P,T,W
AVT			84	4	4.8	1 305 680	235	0.020	SCC/IGA,P,D,E,T
AVT/CD	Boric acid	1982	1	1	100	6 776	419	6.2	P
AVT + CD	Phosphate	1974	4	2	50	36 598	13	0.035	SCC/IGA,D,T
AVT + CD			43	6	14	721 032	658	0.091	SCC/IGA,E
Phosphate			11	4	36.4	119 149	157	0.134	W
Boric acid	AVT	1980 to 1985	8	3	37.5	100 513	298	0.296	P,T,D,SCC/IGA

[a]Includes only corrosion from the secondary side.
[b]Abbreviations used are

AVT	All-volatile treatment	P	Pitting
CD	Condensate demineralization	W	Wastage
SCC	Stress corrosion cracking	E	Erosion
IGA	Intergranular attack	T	Thinning
D	Denting		

Table 7 Experience with Steam Generator Tube Materials in 1985

Tube material	Number of reactors	Number of tubes	Number of tubes plugged	Tubes plugged, %	Assigned failure mechanism[a]
Alloy 600	136	1 866 326	6 573	0.352	SCC/IGA(OD), SCC(ID),W,D,Fr,P,F,E
Alloy 800	17	227 286	156	0.069	W
Stainless steel	5	86 208	108	0.125	SCC/IGA(OD),W
Monel 400	10	256 758			

[a]SCC	Stress corrosion cracking	F	Fatigue
IGA	Intergranular attack	P	Pitting
W	Phosphate wastage	E	Erosion
D	Denting	(OD)	Secondary side
Fr	Fretting	(ID)	Primary side

the plugged tubes were at the Biblis stations. All the stations plugging Alloy 800 tubes as a result of wastage used phosphate chemistry control. Only 5 reactors use stainless steel steam generator tubes, and 108 of these tubes were plugged in 1985, all at Yankee Rowe. Corrosion of the stainless steel tubes was mostly wastage and SCC/IGA at the tubesheet.

None of the alternative materials (to Alloy 600) have shown SCC(ID); some of the reactors have been in operation more than 15 yr. Also, reactors with thermally treated Alloy 600 tubes have not shown this defect mechanism to the end of 1985.

STEAM GENERATOR TUBE REPAIRS

When tube defects occur within or above the tubesheet, it is possible to sleeve the tube with new material and thereby preserve the heat-transfer area. Sleeving has been successful at many reactors, ranging from a few tubes to prove the process to more than half of all unplugged tubes. At Beznau 1, previously plugged tubes have been returned to service after sleeving.

In a previous review, we provided an exhaustive list of tube-sleeving operations.[13] Table 8 shows reactors at which sleeving was conducted during 1985. The most common defects for which sleeving repairs were made are SCC(OD)/IGA and pitting.

Table 8 Tube Repairs by Sleeving During 1985

Reactor name	Tubes sleeved	Comments
Beznau 1	79	SCC/IGA[a]
Beznau 2	39	SCC/IGA
Genkai 1	147	SCC/IGA
Ginna	69	SCC/IGA
Indian Point 3	635	Pitting
Ohi 1	451	SCC(ID), SCC/IGA
Ringhals 2	59	SCC/IGA

[a]Stress corrosion cracking (SCC) or intergranular attack (IGA).

In previous years, SCC(ID) and wastage defects have also been repaired. More than 19 800 sleeves had been installed by the end of 1985.

There is increasing concern with SCC(ID) at the roll-transition region in the tubesheet. During 1985, 24 reactors were affected and 613 tubes were plugged because of Coriou-type cracking[16] in this highly stressed area. Of several possible remedies for susceptible steam generators, shot peening and roto peening have been widely applied. Shot peening is performed with micropellets of materials, such as Alloy 600, that are projected against the inside surface of the tube in the highly stressed area. Roto peening uses tungsten carbide pellets of approximately 1 mm in diameter bonded to a flap wheel.[20] Both methods impart compressive stresses to the tube surface to counteract the residual stresses that resulted from the tube rolling operation conducted during fabrication. Table 9 lists the reactors at which shot and roto peening were conducted to the end of 1985. It is expected that the hot legs of tubes at all susceptible reactors will be treated. Some steam generators will also be treated on the cold leg. Steam generators with hydraulically or explosively bonded tubes are thought to have lower residual stresses. Also, steam generators with thermally treated Alloy 600 tubes (e.g., 705°C for 10 h) and those with tubes of other materials are not thought to be susceptible. However, kiss rolling, or application of a partial roll above the hard roll, has proven to be unsuccessful in preventing cracks or arresting their growth.[21]

Another area susceptible to SCC(ID) is the apex of tubes with small-radius U-bends. Heat treatment can be applied to some of these tubes. However, no operating reactors had U-bend heat treatments until 1986.

Most steam generators are designed with a 10 to 20% excess heat-transfer area to accommodate occasional plugging and fouling. When this reserve is consumed, it is necessary to derate the reactor or replace the steam generators. To the end of 1985, steam generators had been replaced at seven units. No reactors were shut down during 1985 for steam generator replacement. However, planning for replacement proceeded at Indian Point Station Unit 3, Ringhals 2, Millstone Nuclear Power Station Unit 2, and Donald C. Cook Nuclear Power Plant Unit 2. The replacement steam generators incorporate many features to help minimize operational problems. These include more open tube supports and more corrosion-resistant materials for tubing and support structures. Replacement steam generators have now operated for periods ranging from 300 to 1400 EFPD with no significant problems. For instance, only four tubes have been plugged at Surry 1. Surry 2 has now generated more power with the replacement steam generators than with the original units. Because of the long lead times for construction of nuclear power reactors, improvements developed from research and operational experience can only be introduced slowly into new plants. Replacement steam generators provide an opportunity to gain experience with design improvements more quickly. The experience to date is very encouraging. This is graphically illustrated in Fig. 5, which compares the experience of the two sets of Surry 2 steam generators.

Table 9 Steam Generator Tubes Treated by Shot Peening or Roto Peening to December 31, 1985

Reactor name	Type of treatment	Extent of treatment	Approximate date
Doel 4	Roto peen	Hot leg of all tubes (preoperational)	November 1984
Tihange 3	Roto peen	Hot and cold legs of all tubes (preoperational)	February 1985
Doel 3	Shot peen	Hot legs of all tubes	July 1985
Bugey 5	Shot peen Roto peen	300 hot leg tubes as test Similar to shot-peen test	September 1985
Almaraz 1	Shot peen	Hot legs of all tubes	December 1985
Ringhals 2	Shot peen	Hot legs of all tubes	December 1985
Summer 1	Roto peen	One half of hot legs	December 1985

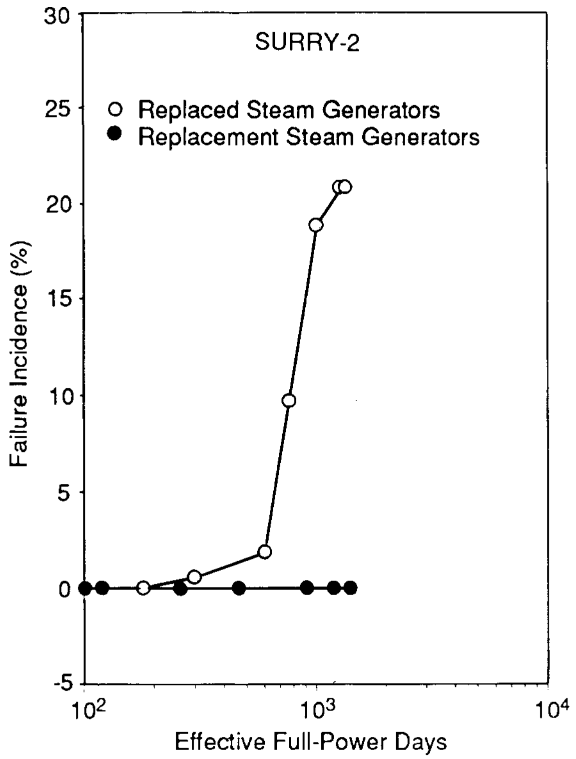

Fig. 5 Comparison of the performance of original and replacement steam generators at Surry 2.

TUBESHEET SLUDGE DEPOSITS

Data regarding steam generator sludge deposits were available for 88 reactors. Some indication of the consistency of the sludge was available for 52 reactors. It was described as soft at 30 reactors, hard at 7 reactors, granular at 7 reactors, and having both hard and soft components at 8 reactors. There was no obvious effect of cooling water salinity on the consistency. However, most reactors using volatile treatment had soft sludge. All reactors using phosphate treatment had sludge containing both hard and soft components. Reactors using all-volatile treatment and condensate demineralization tended to have sludge consistency in all four categories. The sludge varied in depth up to 457 mm (18 in.).

Chemical analysis was performed on the sludge at 25 reactors. Generally, magnetite was the largest constituent, followed by copper, nickel oxide, and zinc oxide. Substantial amounts of copper were found in sludge at 10 reactors. Copper-rich sludge has been associated with pitting at several reactors, especially those using brackish water or seawater for condenser cooling.

The most common method for removing sludge was by water lancing, which was performed at 48 reactors. Amounts removed, where known, varied between a few kilograms and almost 500 kg.

Over the past few years, the consistency of sludge and the chemical composition have not changed appreciably at reactors for which data are available. Sludge depth has also not changed and in some cases has been reduced slightly by lancing.

IN-SERVICE INSPECTION OF STEAM GENERATORS

Inspection data were available for 91 reactors. At least some form of inspection was performed at 60 plants, whereas no inspection was performed at 31 plants.

Table 10 shows the number of tubes inspected in various regions of the steam generators. Inlet tubesheet areas are inspected for SCC(ID), SCC(OD)/IGA, and wastage. In some cases the inspection is carried through the support plate regions (SCC/IGA) or through the U-bend [fretting, SCC(ID)]. Outlet tubesheet areas are inspected for pitting, and extension through the support plates can detect thinning. At 44 plants the inspection was performed for the full length of the

Table 10 Summary of In-Service Inspections During 1985

	Inlet through the tubesheet	Inlet to the supports	From inlet through the U-bend	Outlet through the tubesheet	Outlet to the supports	Full length	Total
Number of tubes	63 715	55 292	14 967	5 035	39 021	293 692	471 722
Tubes in service, %	2.6	2.3	0.6	0.2	1.6	12.0	19.3
Number of reactors	9	14	13	2	8	44	90
Reactors, %	5.3	8.3	7.7	1.2	4.8	26.2	53.6

tubes. Overall, 19.3% of tubes in service were inspected.

Table 11 shows the inspection techniques used. Multifrequency, remote eddy-current testing was by far the most common inspection technique both by plant and by the number of tubes inspected. Some plants without a history of steam generator tube degradation used manual eddy-current testing on small tube samples. Ultrasonic inspection was used at two reactors to augment the eddy-current testing. Visual inspection was performed on both secondary and primary sides, and hydrostatic testing was used to locate leaking tubes or prove that the steam generators were fit for service.

Table 11 Methods Used to Inspect Steam Generators During 1985

Inspection method	Number of reactors
Automated eddy current	58
Manual eddy current	2
Ultrasonic testing	2
Visual	6
Hydrostatic	6

CONDENSER TUBE MATERIALS

At freshwater sites, admiralty brass remains the most common tube material (Fig. 6). Over time, both admiralty brass and stainless steel have maintained their relative share of condenser tubing, even though the number of reactors increased from 21 in 1972 to 101 in 1985. The other tubing materials include aluminum brass, titanium, 70-30 cupronickel, and 90-10 cupronickel.

Since 1976, 14 freshwater-cooled plants have retubed the condensers, and 6 of these occurred during 1985. The condensers at Pickering 1, 2, and 3 were retubed with admiralty brass tubes. In Europe, however, the trend is to replace admiralty brass condenser tubes with stainless steel or titanium. In the United States, the trend is also to replace with stainless steel tubes for freshwater service.

At brackish water sites, the trend is away from copper alloys and toward more corrosion-resistant metals (Fig. 7). The most common retubing or new condenser tube material in recent years has been titanium, although alloy AL-6X is also being used. Nine reactors using brackish cooling water have retubed the condensers (47% of reactors on

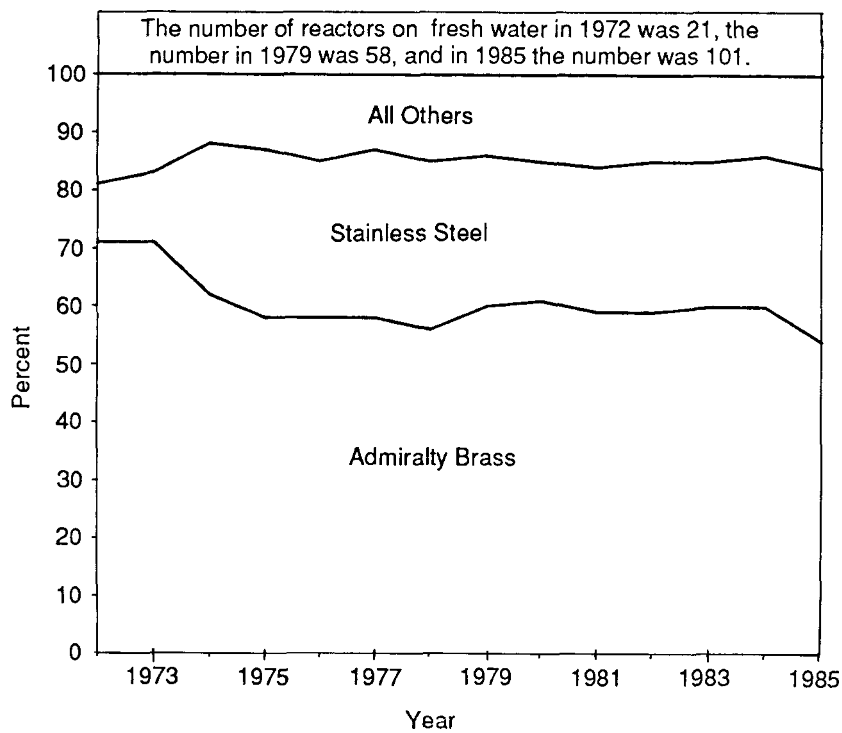

Fig. 6 Condenser tube materials at freshwater sites.

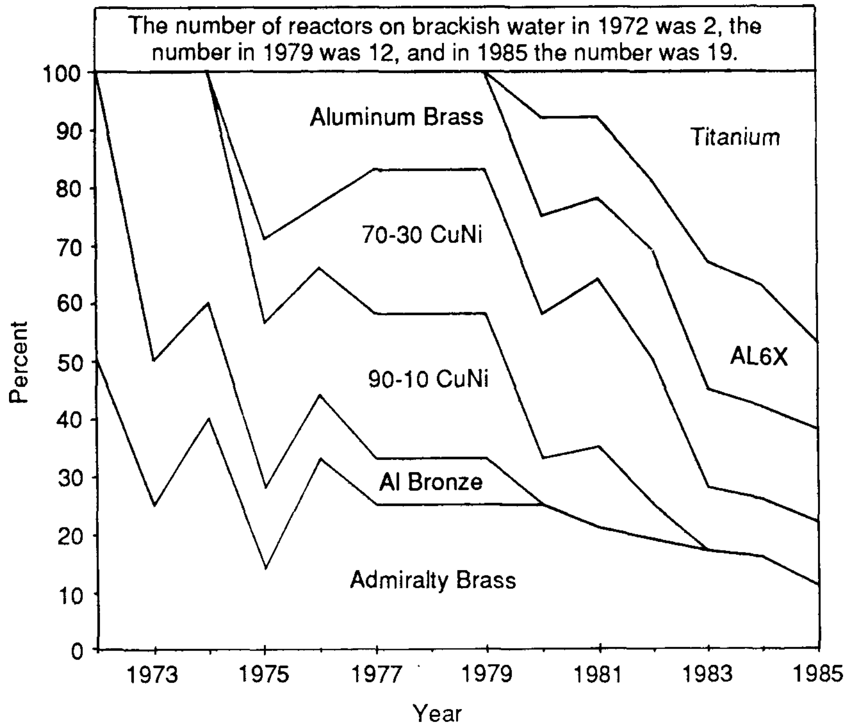

Fig. 7 Condenser tube materials at brackish water sites.

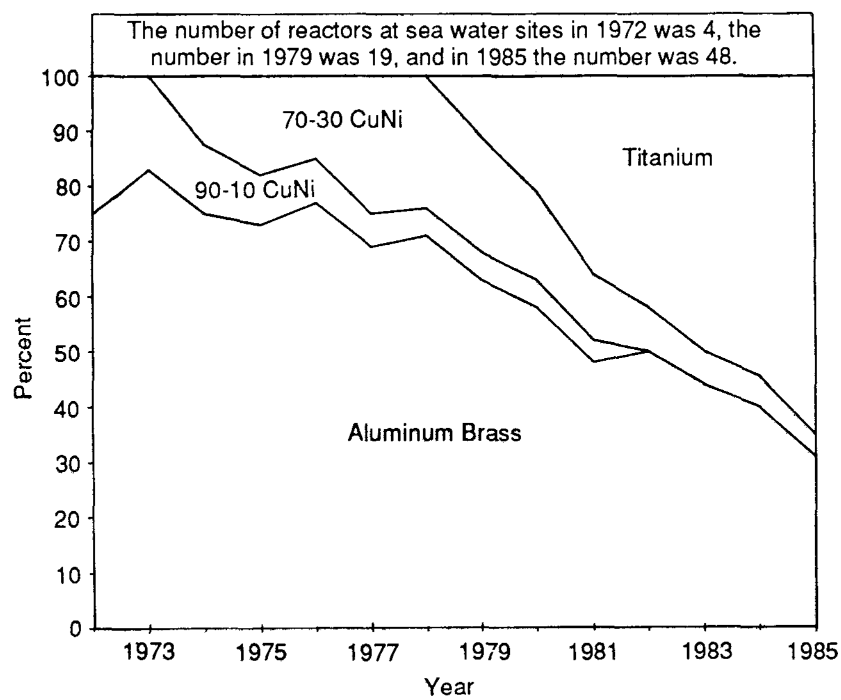

Fig. 8 Condenser tube materials at seawater sites.

brackish water). All of these have converted to titanium or AL-6X. Maine Yankee Atomic Power Plant has now changed condenser tube material twice, from aluminum brass to AL-6X to titanium, following corrosion problems with AL-6X tubing.

At seawater sites, the trend is also toward titanium tubing (Fig. 8). Most of the plants with aluminum brass are the older ones in Japan. By rigorous attention to inspection and repair, these condensers have experienced leak-free service for many years. However, the newer Japanese plants, as well as newer plants everywhere else, use titanium condenser tubes. Eight seawater-cooled plants have retubed the condensers, but none did so in 1985.

Between 1976 and 1986, 31 reactors retubed the condensers, and some did so twice.

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Systems Interaction Analyses: Concepts and Techniques (Part II)

By M. D. Muhlheim[a] and G. A. Murphy[b]

Abstract: *The issue of systems interaction events in nuclear power plants was originally raised because of the concern about certain types of dependent failures in nuclear power-plant safety systems. The source of this concern is that the interdisciplinary review process required in designing, constructing, and operating nuclear power-plant systems may not be adequate to ensure that redundant safety systems are truly independent. In fact, some operating events have demonstrated that the current process may not adequately ensure that failures caused by systems interactions do not occur. Part I of this article reviewed three of the four qualitative analysis techniques that can be used to identify possible systems interactions. Part II of this article reviews the last analysis technique that can be used to identify systems interactions. Each technique, by itself, cannot adequately identify all three types of systems interactions: functional, spatial, and induced-human. Some combination of these techniques is required to perform adequately a systems interaction study or to be incorporated into a probabilistic risk analysis.*

This is the second part of a two-part series discussing several methodologies that can be used to identify possible systems interactions. Part I of "Systems Interaction Analysis: Concepts and Techniques" [see *Nucl. Saf.*, 30(2): 252-265] provided the definition of an adverse systems interaction (ASI) and discussed three of the four methodologies that can be incorporated into a probabilistic risk analysis (PRA) to identify systems interactions. The types of methodologies reviewed in Part I were operating experience reviews, onsite inspections, and analysis-by-parts techniques.[1] Part II discusses graph-based analyses. As in Part I, a simplified flow diagram of the residual heat removal (RHR) system (Fig. 1) will be used to provide the examples (if applicable). This simplified system consists of two parallel trains to inject coolant into the reactor. The trains can take suction from the sump, a reactor coolant system (RCS) hot leg, or the refueling water storage tank, depending on the mode of operation. However, this example evaluates the RHR alignment to the RCS hot leg only. So that the example can be kept simple, heat-exchanger failures or check-valve failures will not be considered, and only one leg (Leg 1) of the RHR system is addressed in the logic models (see Fig. 1). Therefore the only component failures that will be considered in this example are the inlet valves from the suction line (labeled V1A and V1B), the RHR pump, and the isolation valves to the RCS cold legs (labeled V2 and V3). For the example, the system is assumed to be operating (i.e., the valves are open and the pump is running).

GRAPH-BASED ANALYSES

The last class of analysis techniques, graph-based analyses, is comprehensive within a given set of boundary conditions and is used to represent the logical relationships among those components (or systems) whose failures can lead to a specific undesired event. These relationships are captured in the graphic model, and all the potential failure modes (within the scope of the analysis) are then identified from the model by generating (often through the use of computers) the combinations of component and human failures that contribute to the undesired event.

Advantages of this class of techniques include the ability to (1) cover low-frequency events systematically, (2) deal with complex systems, (3) evaluate shared support systems, and (4) identify common-cause failures. Disadvantages of these techniques, when performed at a detailed level, include their complexity and expense.

Six graph-based analysis techniques are reviewed: event tree analysis, fault tree analysis, cause–consequence analysis, digraph matrix analysis (DMA), GO methodology, generic analysis, and sneak-circuit analysis.

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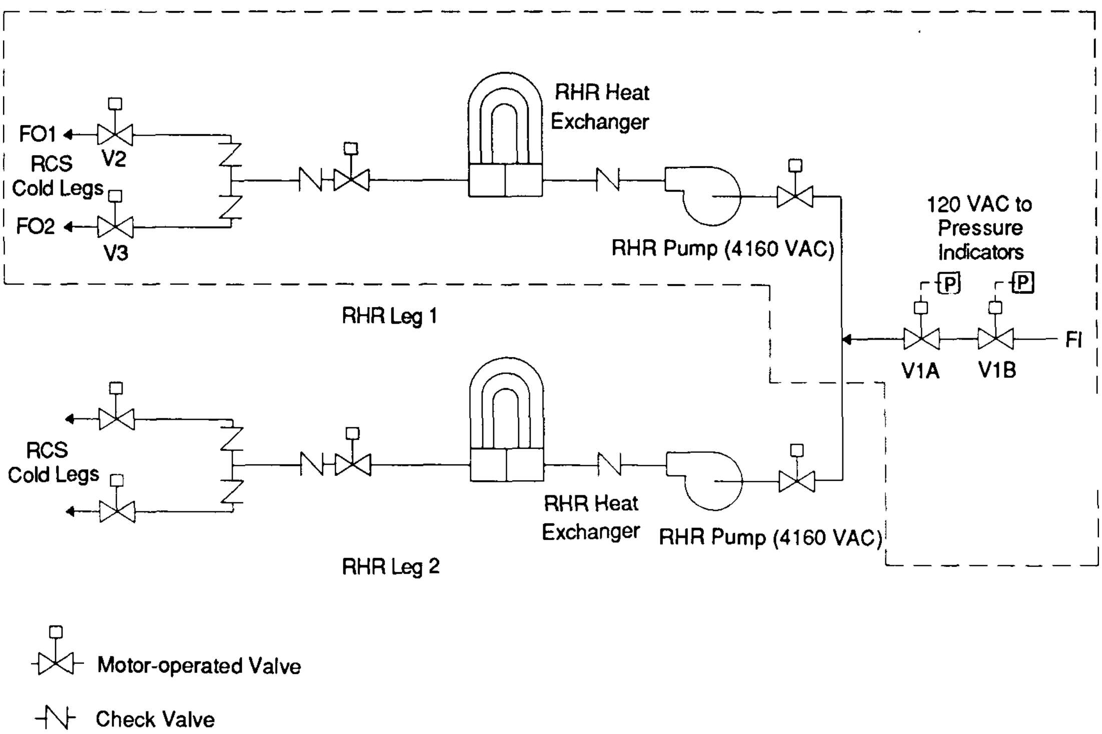

Fig. 1 Simplified residual heat removal (RHR) system flow diagram.

Event Tree Analysis

Because nuclear power-plant systems are so complex, it is generally not feasible to identify by inspection a listing of all important accident sequences. Therefore a systematic and orderly approach is required to understand properly and identify the many factors that could influence the course of potential accidents. This approach involves developing an event tree, which is an inductive logic model that sequentially models the progression of events (both failure and success) from some initiating event to a series of logical consequences. An event tree begins with an initiating failure and maps out a sequence of events (typically on the system level) that forms a set of branches (Fig. 2). Each branch represents a specific sequence.

Event trees are normally used to model events having binary failure states and usually corresponding to total success or failure of a system. For example, the top branch in Fig. 2 ($A\overline{B}\overline{C}\overline{D}\overline{E}$) represents the sequence [with the initiating event (A) being a loss-of-coolant accident (LOCA)]:

- V1A and V1B remain open ($\overline{B}$)
- The pump continues to run ($\overline{C}$)
- V2 remains open ($\overline{D}$)
- V3 remains open ($\overline{E}$)

Therefore this top branch represents the sequence 100% flow through this RHR leg to the RCS cold leg. From this example we see that a complete event tree analysis requires the identification of all possible initiating events and the development of an event tree for each.[2–4] (In many cases, the event trees are identical for several different initiating events, and a single tree is used to represent all these events.)

From the event tree in Fig. 2, it is shown that, if valve V1A *or* V1B transfers closed, or if the pump fails off, or if V2 *and* V3 transfer closed, there will be no flow from RHR Leg 1 to the RCS cold legs. (Obviously, if V1A *or* V1B transfers closed, there will be no flow from either RHR leg. If the pump fails off or if V2 *and* V3 transfer closed, there may still be flow to the RCS cold leg through RHR Leg 2.) Therefore this event tree

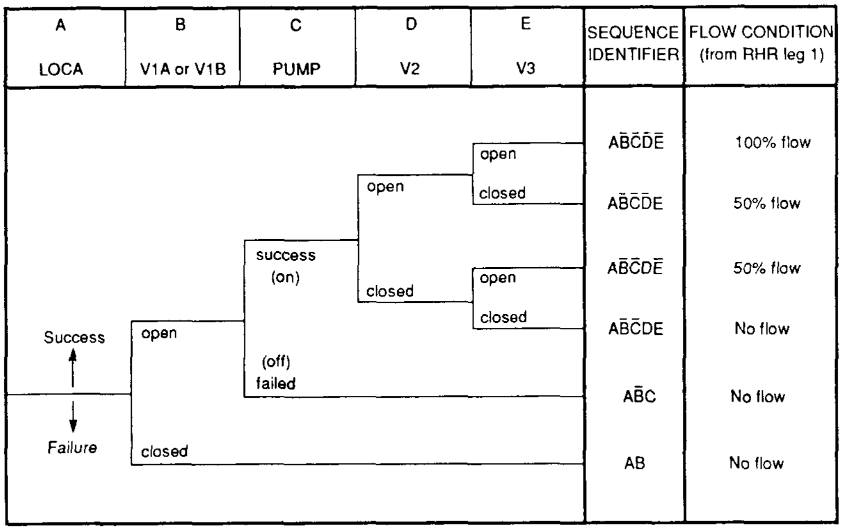

Fig. 2 Event tree for leg 1 of residual heat removal system.

has identified components (or event sequences) that should be evaluated in further detail (i.e., in addition to basic failures, the support system failures or operator actions that can cause the valves to transfer closed or the pump to transfer off).

Fault Tree Analysis

Fault tree analysis is a deductive failure analysis that focuses on an undesired event and provides a method for determining causes of this event. The undesired event constitutes the top event in a fault tree diagram. Careful choice of the top event is important to the success of the analysis. A fault tree analysis describes an undesired state (usually critical from a safety viewpoint) of the plant or system and analyzes the plant or system to find all credible ways in which the undesired event can occur. The fault tree itself is a graphic model of the combinations of faults that will result in the occurrence of the undesired event. The faults can depict hardware failure, human error, system failures, external events (e.g., earthquakes or internal fires), or other events that can lead to the undesired event.[2–4] Therefore a fault tree can be used to identify functional, spatial, and induced-human interactions.

A fault tree is not a model of all possible plant or system failures or all possible causes for failure. Rather, a fault tree is tailored to its top event and includes only those faults which contribute to that top event. The fault tree itself is not quantitative; however, the results can be evaluated quantitatively. In fact, the fault tree is a convenient model to quantify and, along with event trees, has formed the structure for most of the PRA studies carried out for the nuclear industry.[5–6] As a result, many personnel in the industry are experienced in developing and/or using fault trees.

The fault tree for our example is shown in Fig. 3. As can be seen from this figure, fault trees logically describe the relationship between component failures and the top event. Fault trees represent component failures as well as the support system failures (i.e., those systems which provide support to the component of interest). The fault tree logic for the support systems can be developed further and reveal dependencies on additional systems.[7] (For simplicity, the failure logic for the support systems was not developed further in Fig. 3. For example, rather than developing the fault logic for the loss of the 4160-V a-c power bus that provides power to the RHR pump, the undeveloped event "loss of 4160-V a-c power to the RHR pump" was used.)

The minimal cut sets (MCSs) from the fault tree (Fig. 3) are listed in Table 1. The MCSs represent the failure modes of the system. This table shows that component failures in systems that provide support to the RHR can cause the RHR to fail. An analysis of a more realistic problem (and hence a more complicated problem) will

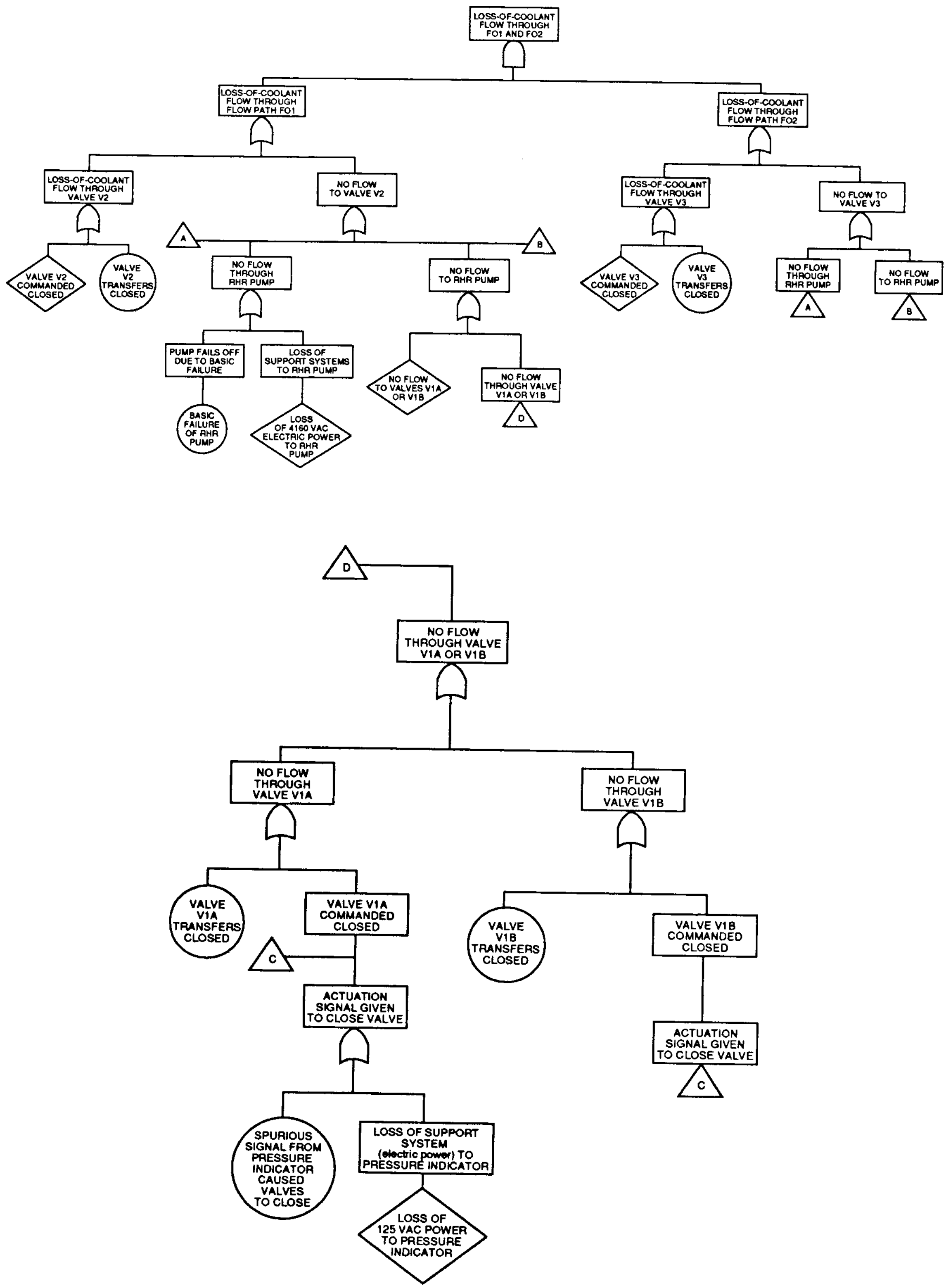

Fig. 3 Fault tree for leg 1 of residual heat removal system.

Table 1 MCSs Identified from a Systems Interaction Fault Tree

MCS No.	Component failure	Resulting interaction
1	Valve V1A transfers closed	
2	Valve V1B transfers closed	
3	RHR pump fails to run	
4	Valve V2 transfers closed Valve V3 transfers closed	
5	Valve V2 commanded to close Valve V3 commanded to close	
6	Valve V2 transfers closed Valve V3 commanded to close	
7	Valve V2 commanded to close Valve V3 transfers closed	
8	Loss of 125-V a-c power to pressure indicator	Causes valves V1A and V1B to transfer closed
9	Spurious actuation signal from pressure indicator	Causes valves V1A and V1B to transfer closed
10	Loss of 4160-V (a-c) electric power to the RHR pump	Causes RHR pump to fail off

show that functional systems failures can combine with support systems failures to cause the occurrence of the event of interest (i.e., the top event). [The logic between the support system and the active component it supports is "OR" logic. For example, the pump can fail because of a basic failure *or* the pump can fail because of a loss of electric power (one of its support systems).]

Cause–Consequence Analysis

A formalized combination of event tree and fault tree analyses is called a cause–consequence analysis.[3] The event trees are used to determine the sequence of events that can lead to the consequences of interest. Event trees are developed for several different initiating events (usually LOCAs and transients). The fault trees are then used to model the causes of the event sequences. The causes of the event sequence failures can be modeled as system failures or component failures. However, if a lack of failure data exists on the system level, the causes would be modeled on the component level, where such data are more readily available. Thus the results of a cause–consequence analysis could be both qualitative and quantitative.

Brookhaven National Laboratory performed a systems interaction analysis on the Indian Point 3 power reactor using a cause–consequence analysis (i.e., fault trees and event trees) with the aid of a failure modes and effects analysis (FMEA).[8] This study identified a previously unrecognized single failure point of the low-pressure injection system: the failure of a battery that powers a 125-V (d-c) vital bus coupled with the initiating event "large or medium LOCA prevents dc control power from being available to automatically actuate the low-pressure injection pumps."

Digraph Matrix Analysis

Similar to the fault tree, a DMA uses "AND" and "OR" gates to model explicitly the logical relationships between components. However, unlike the fault tree, the digraph models the relationships required for *successful* system operation rather than system failure. The DMA uses a success tree that includes all systems and/or components (elements) involved in an accident sequence. This success tree includes subsystems and support systems as elements. A binary matrix that contains information about the relationship between these elements is produced from the success tree. The singletons (one-event MCSs), doubletons (two-event MCSs), etc., can then be identified by using a computer code. Once the MCSs are obtained from the dual digraph, they can be evaluated to identify functional, spatial, and induced-human interactions.[9-10]

When constructing a digraph matrix, the analyst must first select the systems of interest for a detailed evaluation (this is equivalent to the PRA event tree analysis designed to find accident sequences). Next, the analyst constructs a single-digraph model for each accident sequence. This graphic approach allows the analyst to develop a binary matrix of elements that have direct influence on an element of higher order.

For the construction of the digraph for the example problem, the success criteria must be identified and modeled. For this problem, the pump must run, valves V1A *and* V1B must be open, and valve V2 *or* V3 must be open. Next, all components that are *directly* necessary for system operation must be identified. These components are then represented by nodes in a graph. The nodes are represented by a circle, and the arrows on the edges between the nodes represent the components in the system. These arrows indicate the direction of flow or propagation of the effect of information, physical movement, power, etc. The digraph thus contains all the components directly responsible for successful system operation along with the logical relationships required for functioning.[11]

The digraph for our example problem is shown in Fig. 4. In this example, the pump will fail if the supply of water fails (FI) or if valve V1A or V1B fails (transfers closed). Thus an "OR" gate joins these components together. The injection into the RCS will fail if there is no flow from both flow paths F01 and F02; thus these flow paths are connected to the RCS by an "AND" gate.

Each component of the system digraph is now expanded by the use of a unit model.[11] This expansion procedure identifies auxiliary component operation that may affect system operation. Therefore the basic digraph is expanded by replacing each component with a unit model for that component. These unit models describe the direct dependence of a component on other components, and thus their inclusion in the system digraph will allow the analyst to uncover additional failures that are introduced by support components. A typical unit model for an active component includes power, control, lubrication, and maintenance input. New components that are identified by the unit model expansion procedure now become the center for continued unit model expansion. For example, power could be expanded to include each transmission line, switch, relay, transformer, etc. As this expansion proceeds, components, locations, operators, and maintenance shared by systems will be discovered. Figure 5 shows the expanded digraph for the example problem.

The connectivity of a network can be represented as a graph (Fig. 5) or completely as an adjacency matrix (Table 2). The rules that define an adjacency matrix are as follows:[11]

$$a_{ij} = 1 \text{ if node i and j are directly connected}$$
$$= 0 \text{ otherwise}$$

The adjacency matrix can be viewed as describing the possibility of flow from node i to node j. The connectivity between all pairs of nodes in a network is contained in the reachability matrix. The determination of whether any arbitrary node is reachable from any other node can be made by Boolean manipulation of the adjacency matrix (the result is the reachability matrix). Consequently all nodes can be easily represented in the conditional adjacency matrix format by combinations of AND and OR gates. The computer codes CLAMOR[12] and SQUEAK[13] can be used to find reachability sets of any order that reach any node. These reachability sets are the equivalent to MCSs. For this simple example, the reachability sets are identical to those MCSs identified by fault tree analysis (Table 1).

Some advantages of a DMA include the following: (1) the construction of the logic model is performed directly from plant schematics (e.g., piping and instrumentation diagrams, electrical schematics, and safety logic diagrams) so that the model can be readily understood, reviewed, and corrected; (2) the digraph can represent physical situations that are cyclic; (3) the binary matrix indicates all levels of subordination, but only direct first-level relationships must be provided because the applicable computer codes deduce any consequent levels of subordination; and (4) an element of the matrix can be any entity of interest (e.g., an entire system, a system function, component, or maintenance crew), and elements of any level of detail can be intermixed.

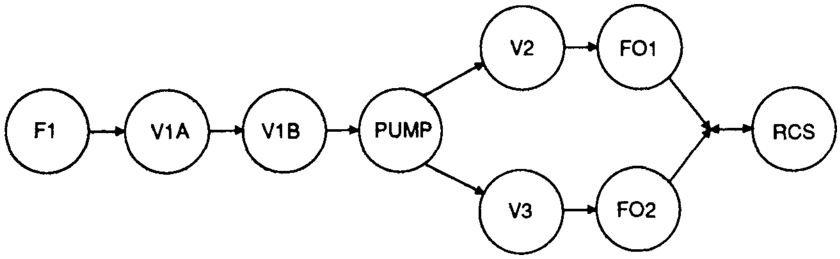

Fig. 4 Basic digraph matrix for the example problem (see text).

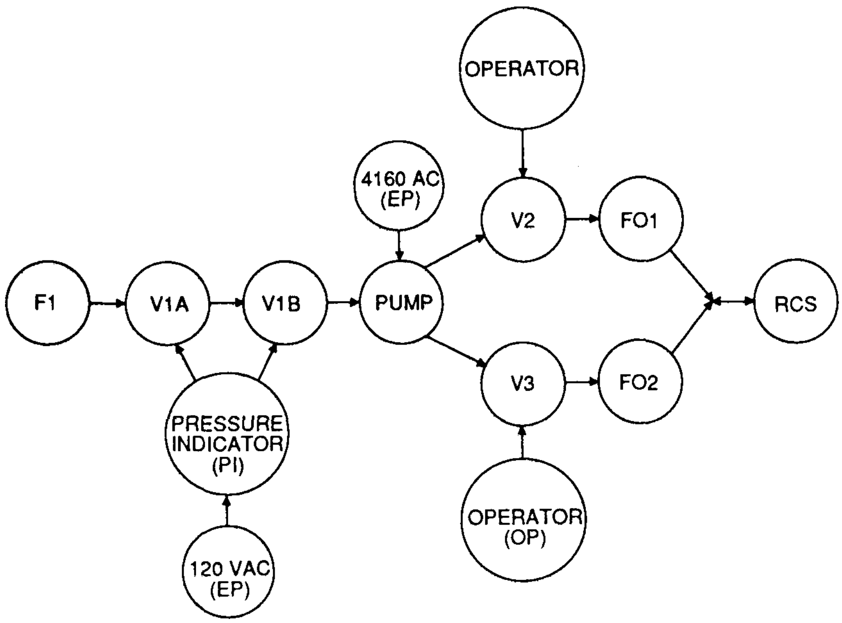

Fig. 5 Expanded digraph matrix for the example problem (see text).

Table 2 Adjacency Matrix for Sample Problem

		V1A	V1B	Pump	V2	V3	FI	F01	F02	EP	PI	RCS
To:	V1A	0	1	0	0	0	0	0	0	0	0	0
	V1B	0	0	1	0	0	0	0	0	0	0	0
	Pump	0	0	0	1	1	0	0	0	0	0	0
From:	V2	0	0	0	0	0	0	1	0	0	0	0
	V3	0	0	0	0	0	0	0	1	0	0	0
	FI	1	0	0	0	0	0	0	0	0	0	0
	F01	0	0	0	0	0	0	0	0	0	0	1
	F02	0	0	0	0	0	0	0	0	0	0	1
	EP	0	0	1	0	0	0	0	0	0	1	0
	PI	1	1	0	0	0	0	0	0	0	0	0
	OP	0	0	0	1	1	0	0	0	0	0	0

Disadvantages of DMA include a scarcity of trained analysts and, for certain types of logic diagrams, computer limitations because of the analyst's attempt to be more complete.

Lawrence Livermore National Laboratory performed a systems interaction analysis on the Indian Point 3 power reactor using DMA (Ref. 14). This study identified a previously unrecognized two-event, core-melt cut set with an estimated frequency of 10^{-4}/year; other two-event MCSs were also identified, but they did not have an impact on the quantitative results of the PRA. The MCS that did impact the quantitative results of the PRA was the transfer close of a service-water valve coupled with the loss of offsite power. In addition, this study showed that the use of digraph matrices is a viable alternative to fault trees for performing a PRA and for identifying systems interactions.

GO Methodology

The GO methodology is a success-oriented technique generally used for quantitative analyses. However, this methodology can be used to construct event trees and to identify component failure combinations that can lead to system failure. Completed GO models resemble system schematics or process flowcharts and tend to be more compact than equivalent fault tree models (though with correspondingly less failure mode information). The GO methodology is based on decision tree theory and uses a system flowchart approach to system modeling. A system schematic, or process and instrumentation diagram (P&ID), is "translated" into a GO chart by the substitution of the GO symbols (or operator types) to represent the functions of the components (e.g., valves, pumps, etc.). The components are logically combined with a "success approach" that defines the possible ways in which the system can work and results in a restated flow diagram. Figure 6 shows the operators and provides a brief description of each.[15]

The GO methodology is a structured process for analyzing problems. Once the problem has been defined, the system boundaries identified, and the success criteria established, the GO model can be developed. There are six basic steps in developing a GO model. These steps for our example problem are shown graphically in Fig. 7 and are described as follows:

Step 1: Systems Definition. Definition is simply the collection of necessary information to perform the analysis. Information needed includes system descriptions, schematics, P&IDs, logic diagrams, and operating procedures. Shown is a simple flow diagram with support systems for the example problem.

Step 2: Establish Inputs/Outputs. Every GO model begins with at least one input and often has many inputs that represent the interfacing systems. The output of the model is determined by the success criteria selected previously. The inputs in our example are the suction to the pump, electric (a-c) power to the pump, and pressure indicators to valves V1A and V1B. The outputs are flow through F01 and F02 (i.e., flow through valves V2 and V3).

Step 3: Draw Functional GO Chart. The system is first represented by the proper selection of GO symbols. Independent components are represented by triangle symbols, and dependent components are represented by circle symbols. These symbols are used to represent the system drawing.

Step 4: Define Operator Types. The operation of each component is analyzed, and the GO operator "Type" that most closely represents operation of that physical component is selected. The Type number is the first of two numbers within each GO symbol. In our example, all inputs are shown as Type 5 (triangular operator symbols) because this operator has only one signal or output. The pump and motor-operated valves V1A and V1B are Type 6 because they require power and actuation (they are normally open and fail open on loss of power). The simplest operator to use for valves V2 and V3 is Type 1. The last operator represents the success criteria for the system—flow is passed through valve V2 or through valve V3 (if 100% flow is required, this would be an AND gate). A logical OR gate is a Type 2 symbol, a logical AND gate is a Type 10 symbol, and a function operator is a Type 9 symbol.

Step 5: Define Signal Sequence. The paths between GO operators are called *signals;* signal numbers are arbitrarily assigned to identify the input/output relationship between the GO operators. That is, these numbers connect the GO operators to each other.

Step 6: Define Operator Kinds. The operator "Kinds" are the set of probability data of success-

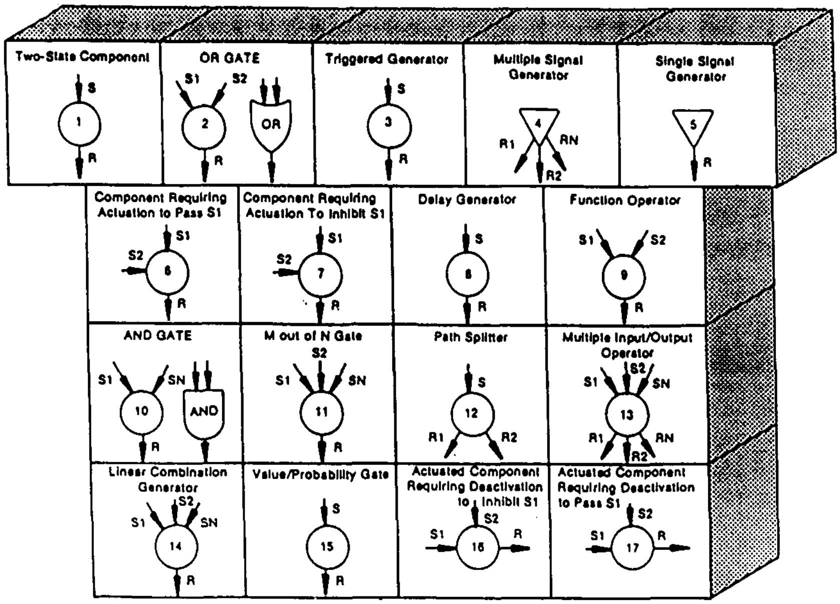

Fig. 6 GO operator types (from Ref. 15).

ful and unsuccessful operation associated with each component. The Kind number is the second of two numbers within each GO symbol. The same Kind numbers are assigned to like inputs and components. Thus, even though the pump and the motor-operated valves are the same Type, they are not the same Kind. However, the motor-operated valves (V1A and V1B) are the same Type *and* the same Kind. The resulting GO model is shown in this step.

The next step in the process is to translate the constructed GO model into an input listing for the GO codes and to execute the codes. From these models and by using the Fault Finder (FF) sequence in the GO code, the dominant contributors and the MCSs to system failure can be identified. The cut sets determined by this GO model are the same as those determined by fault tree analysis (see Table 1).

Specific advantages of the GO methodology are that (1) the system models follow the normal process flow (as does a DMA), (2) modeling of most component and systems interactions and dependencies is explicit, (3) models are compact and easy to validate, (4) model evaluations can represent both success and failure states of systems, and (5) it is uniquely adaptable to analyses in which many levels of system availability are to be considered because it has the ability to handle multiple system states (i.e., partial failure or degraded conditions can be modeled).

Disadvantages of the GO methodology are that (1) fewer analysts are familiar with the GO methodology than with fault tree or event tree analyses, (2) the GO methodology has been used for probabilistic studies of individual systems but has not been used to any great extent as the primary technique for a full-scope PRA, and (3) the MCS size is limited to four events and the number of minimal cut sets is limited to 4000.

Generic Analysis

A generic analysis reviews the basic events in each MCS for susceptibilities to generic causes of failure (dependencies). The MCSs can be determined from fault trees or similar analyses. When a generic cause of component failure is common to all members of an MCS, the MCS is called a common-cause candidate. Generic causes for failure that are often considered in such analyses are listed in Table 3 (Ref. 16).

A generic analysis is a helpful, methodical way to identify spatial systems interactions. It has been implemented in a number of computer programs and has been used in dependent failure analyses in the nuclear industry.

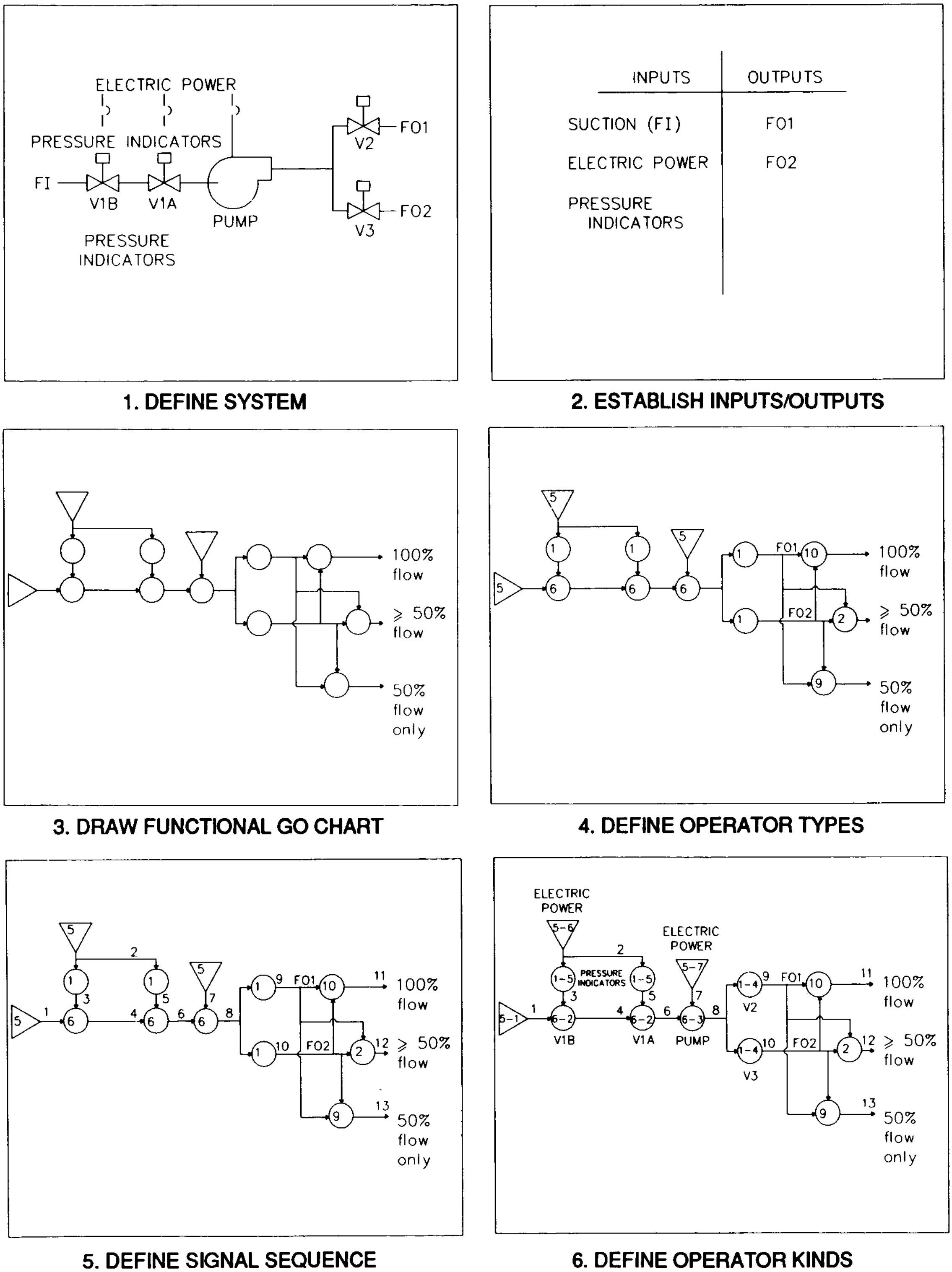

Fig. 7 GO model for example problem (see text).

Table 3 Generic Common-Cause Candidates

Mechanical/thermal generic causes	
	Impact
	Vibration
	Pressure
	Grit
	Moisture
	Stress
	Temperature
	Freezing
Electrical/radiation generic causes	
	Electromagnetic interference
	Radiation damage
	Conducting medium
	Out-of-tolerance voltage
	Out-of-tolerance current
Chemical/miscellaneous generic causes	
	Corrosion (acid)
	Corrosion (oxidation)
	Other chemical reactions
	Carbonization
	Biological
Other common links	
	Energy source
	Calibration
	Installations
	Maintenance
	Operator or operation
	Proximity
	Test procedure
	Energy flow paths

For our example problem, the MCSs for the requirement of 100% flow (without considering support system failures) are

- V1A transfers closed
- V1B transfers closed
- Pump fails to run
- V2 transfers closed *and* V3 transfers closed

Assume that the susceptibilities to valves V2 and V3 involve a common maintenance crew. (See the fourth item under other common links in Table 3.) Improper maintenance could cause the failure of valves V2 *and* V3 (e.g., suppose the maintenance crew improperly installed the valve seats on *both* valves). Generic analyses can also be used to evaluate spatial interactions, such as fire, flooding, harsh environments, etc., because of location commonalities (i.e., no barriers separating the components in the same MCS) and human errors, such as operator errors, maintenance errors, etc.

Sneak-Circuit Analysis

A sneak-circuit analysis (SCA) is normally applied to electrical systems or to computer software (an area not analyzed by the other methods). The SCA was originally designed to identify unplanned modes of operation, unexplained problems, and unrepeatable glitches or anomalies. The SCA is usually applied to electromechanical circuits but can be used for discrete analog and digital circuitry.[17] This type of analysis can also be applied to fluid systems because fluid systems can be represented by electrical system analogs (however, only feasibility studies have been undertaken to date).

An SCA will identify latent signal paths or circuit conditions within a system that may cause undesired functions to occur or inhibit a desired function from occurring. The problems identified in the analysis are called sneak circuits and are characterized by their ability to escape detection during most standardized tests. In addition, sneak circuits are not dependent on component failures, although many erroneous responses of system failures occur because of component failures. Sneak circuits can be subdivided into four types:[18]

1. Sneak paths cause current or energy to flow along unexpected paths.
2. Sneak timing may cause or prevent the flow of current or energy to activate or inhibit a function at an unexpected time.
3. Sneak indications may cause an ambiguous or false display of system operating conditions.
4. Sneak labels may cause incorrect stimuli to be initiated through operator error.

An example of a sneak circuit, the "H" pattern circuit, is shown in Fig. 8. Each leg of the H is fed from independent power sources, and the lower circuitry must provide proper isolation. If isolation is not maintained, a bus-to-bus sneak circuit is generated. Two equal power sources can still generate sneak circuits whenever one bus develops an increased or decreased voltage level relative to the second bus. The resulting voltage and current shifts can inadvertently activate components in the H pattern. A short on one bus could short the second bus and thus induce undesired equipment functions and no convenient means or capability to reset the system.

An advantage of SCA is that problems caused by latent signal paths that are not contingent upon

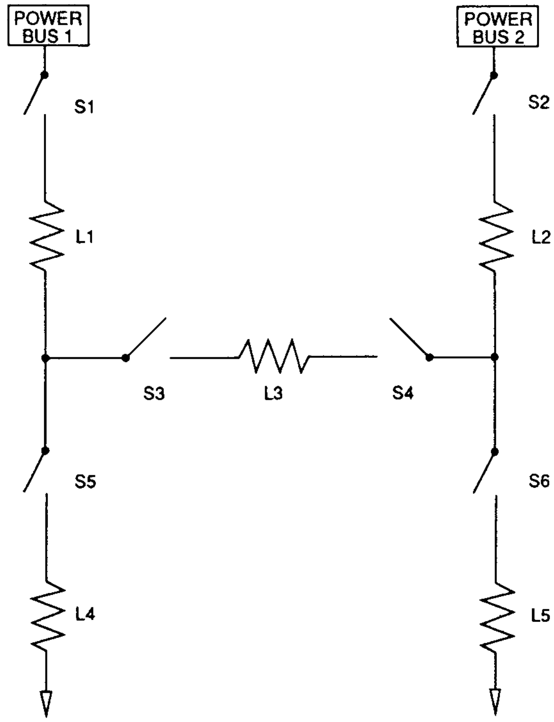

Fig. 8 "H" pattern circuit.

component failures can be identified. These signal paths can cause undesired events to occur or inhibit a desired function from occurring. In addition, SCA can be used to evaluate computer hardware and software, which is an increasingly important aspect of any control system. The main disadvantage of SCA is the lack of qualified analysts able to perform such analyses. In addition, an SCA can require a significant amount of engineering time and thus can be expensive (however, this is consistent with all the graph-based analysis techniques). Consequently it is generally applied to components and circuits that are considered critical.[19]

CONCLUSIONS AND RECOMMENDATIONS

Many different methodologies that can identify systems interactions are available. However, no one methodology by itself can adequately identify functional, spatial, and induced-human interactions. Therefore several different analysis techniques should be used simultaneously. Determining the most appropriate combination of analysis techniques for identifying systems interactions requires consideration of several factors: time, scope, costs, benefits, etc. From a review of the methodologies available, three insights became apparent. First, any systems interaction program should use operating experience reviews, design reviews, and preoperational testing. These three methodologies are already required to be performed, and minimal modifications to the existing programs should be required to identify all three types of systems interactions. Second, expanding the scope of PRAs to include the identification of systems interactions should simplify the problem (with respect to starting an independent evaluation) because the analysts would already be familiar with the systems and their responses. Finally, the resulting combination of methodologies must be able to adequately identify all three types of systems interactions: functional, spatial, and induced-human.

Table 4 presents a possible combination of methods that can be used to identify systems interactions. As stated previously, it is recommended that all systems interaction programs use operating experience reviews, design reviews, and preoperational testing. Some central agency (e.g., Institute of Nuclear Power Operations or Electric Power Research Institute) could evaluate the operating experience reviews performed by Oak Ridge National Laboratory[20,21] and by Fluor Pioneer Inc.[22] and develop a complete list of systems interactions that have been identified (by combining the results of these studies with the results of their own programs) along with the

Table 4 Combinations of Methods that can be Used to Identify Systems Interactions

Group No.	Methodology	Primarily used to identify the following interactions
1	Operating experience reviews Design reviews Preoperational testing	Spatial, functional, and induced-human interactions
2	Plant walk-through and/or generic analysis (requires fault trees)	Spatial interactions
3	Human factors analysis	Induced-human interactions
4	Event tree/fault tree and/or failure nodes and effects analysis or GO methodology or digraph matrix	Functional interactions

specific concerns and information associated with each interaction. Utilities could then evaluate, with minimal cost, the applicability of the events to their plants.

The systems interaction study would be supplemented by using techniques from Groups 2, 3, and 4. When the analysts are working on these groups, the information from Group 1 should be available to provide insights and stimulation.

In summary, the methodologies discussed in this article can be applied to identify systems interactions. The problem in conducting a systems interaction analysis, however, is not so much with methodology as it is with scope and level of detail.

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Reactor Shutdown Experience

Compiled by J. W. Cletcher

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, covering the three-month period of January, February, and March 1989. Cumulative information, since 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 (Jan. 1, 1989) 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 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 (Jan. 1, 1989, 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 1 Reactor Shutdowns by Reactor Type and Percent Power at Shutdown (For U.S. Power Reactors in Commercial Operation at the Start of the Current Three-Month Period Reported for January, February, and March 1989 and Cumulative Since May 1984)

	BWRs (35)				PWRs (71)			
Reactor power (P), %	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[a]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]
0	22	2.42	514	3.28	16	0.96	379	1.27
$0 < P \leq 10$	1	0.11	137	0.87	6	0.36	175	0.59
$10 < P \leq 40$	3	0.33	126	0.80	10	0.60	288	0.96
$40 < P \leq 70$	3	0.33	102	0.65	8	0.48	164	0.56
$70 < P \leq 100$	11	1.21	254	1.62	21	1.26	369	1.23
100	14	1.54	166	1.06	24	1.44	651	2.18
Total	54	5.94	1300	8.30	85	5.10	2026	6.78

[a]Based on 156.65 BWR reactor years.
[b]Based on 298.94 PWR reactor years.

Table 2 Reactor Shutdowns by Reactor Type and Shutdown Type (For U.S. Power Reactors in Commercial Operation at the Start of the Current Three-Month Period Reported for January, February, and March 1989 and Cumulative Since May 1984)

	BWRs (35)				PWRs (71)			
Shutdown type	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[a]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]
Real scrams	32	3.52	705	4.50	52	3.12	1366	4.57
Spurious scrams	12	1.32	448	2.86	11	0.66	385	1.29
Non-scram shutdowns	10	1.10	157	1.00	22	1.32	274	0.92

[a]Based on 156.65 BWR reactor years.
[b]Based on 298.94 PWR reactor years.

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age[a]
(Reported in January, February, and March 1989 for U.S. Power Reactors Licensed for Full Power and Cumulative Since May 1, 1984. Age Categories Are Shown as of the Start of the Three-Month Period.)

Years in commercial operation (C.O.)	BWRs (35)					PWRs (71)				
	Number		Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year	Number		Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year
	Reactors	Shutdowns				Reactors	Shutdowns			
Not yet in C.O.[b]	0	0	0.00	179	22.60	0	0	0.00	108	16.00
First year of C.O.	1	4	16.00	106	10.96	4	11	11.00	282	14.46
Second through fourth year of C.O.	7	15	8.57	162	7.07	14	22	6.29	416	8.31
Fifth through seventh year of C.O.	4	2	2.00	24	4.36	14	1	0.57	181	5.83
Eighth through tenth year of C.O.	1	1	4.00	92	6.10	6	8	5.33	220	5.22
Eleventh through thirteenth year of C.O.	2	4	8.00	222	5.86	11	13	4.73	363	4.87
Fourteenth year and beyond	20	26	5.20	386	6.71	29	30	4.14	335	4.53

[a]Age is defined as time since start of commercial operation at the beginning of the three-month period.

[b]This category includes reactors licensed for full-power operation but not yet commercial. At the start of the reporting period (Jan. 1, 1989), there were no such units.

Operating U.S. Power Reactors

Compiled by E. G. Silver

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 Mar. 31, 1989, 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 (January, February, and March 1989) covered in this report and the cumulative values of these parameters since the beginning of commercial operation. They are defined as follows:

Unit Capacity (Percent): (Net electrical energy generated during the reporting period × 100) divided by the product (number of hours in the reporting period × Maximum Dependable Capacity).

Forced Outage Rate (Percent): (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 U.S. power reactors during the three months covered by this report (January, February, and March 1989) and for the years 1987 and 1988.

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 space here for routine problems of operation and maintenance, but such information is available at the Nuclear Regulatory Commission (NRC) Public Document Room, 1717 H Street, NW, Washington, D.C. 20555.

Some significant operating events are summarized elsewhere in this section, and a report on activities relating to facilities still in the construction process is given in the article "Status of Power-Reactor Licensing Activities" in the last section of each issue 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

Table 1 Licensed U.S. Power Reactors as of Mar. 31, 1989

Status	No.	Capacity,[a] MW(e) (net)
In commercial operation[b]	108	94 860
In power ascension phase[c]	0	0
Licensed to operate at full power	108	94 860
Licensed for fuel loading and low-power testing[d]	3	3 170

[a]Based on the maximum dependable capacity (MDC) where available; design electrical rating (DER) is used when MDC rating is not available.

[b]Excludes Dresden 1 (DER = 200), Humboldt Bay (DER = 65), Three Mile Island 2 (DER = 906), and La Crosse (DER = 50), all of which have operating licenses but are shut down indefinitely or permanently.

[c]None at this time.

[d]Shoreham (DER = 820), South Texas 2 (DER = 1250), and Vogtle 2 (DER = 1100).

fuel cycle and to management of radioactive wastes in general.

TWO REACTORS CHANGE STATUS

During the first quarter of 1989, two nuclear power plants were licensed for full-power operation. One of them, Vogtle 2, received both low-power and full-power licenses during this time period.

Low- and Full-Power Licenses for Vogtle 2

Vogtle 2, the second of two 1157-MW(e) pressurized-water reactors (PWRs) in Burke County, Ga., is located on the bank of the Savannah River, about 26 miles south-southeast of Augusta, Ga., and 15 miles east-northeast of Waynesboro, Ga. Georgia Power Company and the other owners of the plant (Municipal Electric Authority of Georgia, Oglethorpe Power Corp., and the City of Dalton, Ga.) received a low-power license authorizing fuel loading and testing up to 5% of full power on Feb. 9, 1989 (Ref. 1). Then, on Mar. 31, 1989, the NRC Office of Nuclear Regulation (ONR) authorized full power for this unit,[2] following its sister unit, Vogtle 1, which was licensed for full power almost exactly 2 yr earlier, in March 1987.

Full-Power License for South Texas 2

On Mar. 28, 1989, ONR also issued a full-power license to Houston Lighting & Power Company and the other owners (Central Power & Light Co., and the cities of San Antonio and Austin, Tex.) for South Texas 2, which, with its sister unit, South Texas 1, is located in Matagorda County, Tex., on the Colorado River about 12 miles southwest of Bay City, Tex. (Ref. 3). Like its sister unit, South Texas 2 is a 1250-MW(e) PWR; it received its low-power license on Dec. 16, 1988. Unit 2 follows Unit 1 by only a single year, since South Texas 1 received its full-power license in March 1988.

UTILITIES ASKED TO TAKE STEPS TO PREVENT THERMAL STRATIFICATION DAMAGE

In early 1989 the NRC staff asked all utilities building or operating PWRs to take steps to assure that pressurizer surge lines are not subject to damage as a result of thermal stratification (TS) (Ref. 4). TS is a situation that may occur in horizontal runs of pipe initially filled with cold water if hot water flows into the pipe slowly enough to avoid thorough mixing as a result of turbulent flow. In such an event, the density difference between hot and cold water may lead to a layer of hot water above and cold water below. This situation may occur in the surge line that connects the pressurizer to the primary cooling system.

When a PWR is started up, the water in the pressurizer is heated electrically to form a steam bubble and establish a free surface for pressure control. This causes hot water to flow very slowly through the surge line and thus sets up a situation in which TS may occur, as described previously. The potential problem posed by TS is that the relatively larger thermal expansion of the upper part of the pipe may cause bending stresses that could lead to plastic deformation of the line. For example, the surge line in the Trojan nuclear plant was permanently deformed after it contacted two pipe restraints as a result of the TS phenomenon. In addition, significantly larger-than-expected surge-line displacement has been noted during startup of Beaver Valley 2.

As a result, the staff asked licensees that operate PWRs to:

1. Conduct a visual inspection of the pressurizer surge line the next time the reactor is placed in cold shutdown for more than seven days;

2. Demonstrate, within four months, that the pressurizer surge line meets the applicable design codes and other NRC requirements if the reactor is over 10 years old; owners of reactors in operation less than 10 years are to complete the analyses in one year;

3. Obtain plant specific data on TS if their analyses do not demonstrate compliance with the design codes and other requirements; and

4. Update their stress and fatigue analyses within two years to ensure compliance with applicable code requirements.

The staff also asked licensees building PWRs to:

1. Demonstrate that the pressurizer surge line meets applicable design codes and other NRC requirements before low power testing is authorized;

2. Evaluate operational alternatives or piping modifications needed to reduce fatigue and stresses to acceptable levels;

Table 2 Summary of Operating U.S. Power Reactors as of Mar. 31, 1989[a]

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %				Footnotes
			MW(t)	MW(e)		Jan.	Feb.	Mar.	Cumulative (lifetime)	Jan.	Feb.	Mar.	Cumulative (lifetime)	
ARKANSAS 1 and 2, Pope County, Ark. (Arkansas Power & Light Co.)	50-313 50-368	PWR (B&W) PWR (CE)	2568 2815	850 912	12/74 3/80	63.8 69.5	0 104.0	0 103.6	58.0 67.4	35.8 22.1	100.0 0	96.5 0	14.5 13.8	c
BEAVER VALLEY 1 and 2, Shippingport, Pa. (Duquesne Light Co.)	50-334 50-412	PWR (West) PWR (West)	2652 2660	852 836	10/76 11/87	69.8 90.9	77.1 53.3	86.8 28.2	54.7 82.4	4.5 0	2.3 21.3	0 0	17.8 4.2	
BIG ROCK POINT, Charlevoix County, Mich. (Consumers Power Co.)	50-155	BWR (GE)	240	72	3/63	99.9	96.0	66.3	59.8	0	0	0	13.0	
BRAIDWOOD 1 and 2, Braidwood, Ill. (Commonwealth Edison Co.)	50-456 50-457	PWR (West) PWR (West)	3425 3425	1120 1120	7/88 10/88	54.4 90.2	65.7 30.9	83.2 11.0	76.8 54.6	0 0	0 0	6.5 0	6.8 11.4	c
BROWNS FERRY 1, 2, and 3, Decatur, Ala. (Tennessee Valley Authority)	50-259 50-260 50-296	BWR (GE) BWR (GE) BWR (GE)	3293 3293 3293	1065 1065 1065	8/74 3/75 3/77	0 0 0	0 0 0	0 0 0	39.1 37.4 37.2	100.0 100.0 100.0	100.0 100.0 100.0	100.0 100.0 100.0	47.0 46.7 48.5	m m m
BRUNSWICK 1 and 2, Brunswick County, N. C. (Carolina Power & Light Co.)	50-325 50-324	BWR (GE) BWR (GE)	2436 2436	821 821	3/77 11/75	0 98.3	0 98.5	0 95.3	51.6 49.1	0 0	0 0	0 0	14.5 13.9	e
BYRON 1 and 2, Byron, Ill. (Commonwealth Edison Co.)	50-454 50-455	PWR (West) PWR (West)	3425 3425	1120 1120	9/85 8/87	96.4 4.4	92.2 0	85.0 59.8	64.8 56.6	1.9 0	1.3 0	0 0	4.3 3.9	 e
CALLAWAY 1, Callaway County, Mo. (Union Electric Co.)	50-483	PWR (West)	3411	1171	12/84	104.0	104.7	97.2	79.3	0	0	0	3.6	
CALVERT CLIFFS 1 and 2, Lusby, Md. (Baltimore Gas & Electric Co.)	50-317 50-318	PWR (CE) PWR (CE)	2560 2560	845 845	5/75 4/77	104.0 93.8	101.3 103.9	8.0 48.3	72.9 79.6	5.2 0	1.9 0	50.5 36.5	8.8 5.4	c e
CATAWBA 1 and 2, Lake Wylie, S.C. (Duke Power Co.)	50-413 50-414	PWR (West) PWR (West)	3411 3411	1145 1153	6/85 8/85	0 81.1	60.4 84.7	86.3 27.1	64.0 62.5	0 12.2	8.9 8.6	9.2 0	15.0 21.4	e e
CLINTON 1, Clinton, Ill. (Illinois Power Co.)	50-461	BWR (GE)	2894	933	11/87	1.0	0	0	59.5	0	0	0	5.8	e
COOK 1 and 2, Benton Harbor, Mich. (Indiana & Michigan Electric Co.)	50-315 50-316	PWR (West) PWR (West)	3250 3391	1030 1100	8/75 7/78	73.7 0	68.3 0	27.5 29.6	65.4 58.6	3.4 0	0 0	0 0	7.9 14.5	 e,c
COOPER, Nemaha County, Nebr. (Nebraska Public Power District)	50-298	BWR (GE)	2831	778	7/74	78.4	73.8	94.4	60.3	21.7	15.9	0	4.9	

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, % Jan.	Feb.	Mar.	Cumulative (lifetime)	Forced outage rate, % Jan.	Feb.	Mar.	Cumulative (lifetime)	Footnotes
CRYSTAL RIVER 3, Crystal River, Fla. (Florida Power Corp.)	50-302	PWR (B&W)	2560	825	3/77	33.3	62.4	0	55.9	0	0	0	21.1	c
DAVIS-BESSE 1, Ottawa County, Ohio (Toledo Edison Co.)	50-346	PWR (B&W)	2772	906	7/78	82.2	102.3	84.4	40.5	17.9	0	0	31.6	
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	96.7 101.2	98.6 101.4	98.6 102.1	71.8 71.8	0 0	0 0	0 0	3.3 9.6	
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	0 93.7	10.3 98.0	86.2 77.8	57.8 56.1	0 2.6	13.8 0	7.4 18.7	11.0 12.1	e
DUANE ARNOLD, Cedar Rapids, Iowa (Iowa Electric Light & Power Co.)	50-331	BWR (GE)	1593	538	2/75	27.9	88.9	60.9	52.8	50.1	6.0	5.0	14.4	c
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	100.2 100.5	99.1 94.9	97.7 76.8	70.0 82.3	0 0	0 0	0 0	8.6 4.4	 e
FERMI-2, Newport, Mich. (Detroit Edison Co.)	50-341	BWR (GE)	3292	1093	1/88	55.2	89.4	49.9	49.2	36.8	8.8	37.5	25.9	
FITZPATRICK, Oswego, N. Y. (Power Authority of State of N. Y.)	50-333	BWR (GE)	2436	821	7/75	105.4	106.6	104.9	65.4	0	0	0	10.7	
FORT CALHOUN, Washington County, Nebr. (Omaha Public Power District)	50-285	PWR (CE)	1420	478	6/74	0	71.0	98.1	67.4	0	0	0	2.9	e
FORT ST. VRAIN, Platteville, Colo. (Public Service Co. of Colo.)	50-267	HTGR (GGA)	842	330	7/79	0	0	0	13.4	100.0	100.0	100.0	62.5	c
GINNA, Ontario, N. Y. (Rochester Gas & Electric Corp.)	50-244	PWR (West)	1520	490	7/70	97.8	102.9	51.0	73.7	2.8	0	0	6.1	e
GRAND GULF 1, Port Gibson, Miss. (Mississippi Power & Light Co.)	50-416	BWR (GE)	3833	1250	7/85	104.8	105.9	53.3	69.8	0	0	0	5.7	e
HADDAM NECK, Haddam Neck, Conn. (Connecticut Yankee Atomic Power Co.)	50-213	PWR (West)	1825	582	8/67	104.3	104.4	49.0	78.3	0	0	0	5.7	
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	100.3 98.9	98.8 99.4	99.1 97.9	59.9 62.0	0 0	0 0	0 0	13.3 8.8	

(Table continues on the next page.)

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, % Jan.	Feb.	Mar.	Cumulative (lifetime)	Forced outage rate, % Jan.	Feb.	Mar.	Cumulative (lifetime)	Footnotes
HOPE CREEK, Salem, N. J. (Public Service Electric & Gas Co.)	50-354	BWR (GE)	3293	1067	12/86	101.2	60.5	77.7	79.3	0	0	1.0	6.7	
INDIAN POINT 2 and 3, Buchanan, N. Y. (Unit 2, Consolidated Edison Co. of New York; Unit 3, Power Authority of State of N. Y.)	50-247 50-286	PWR (West) PWR (West)	2758 2760	873 965	8/74 4/76	97.7 98.8	99.9 9.0	46.2 0	61.5 50.6	0 0	1.4 0	1.4 0	8.2 17.8	e
KEWAUNEE, Carlton, Wis. (Wisconsin Public Service Corp.)	50-305	PWR (West)	1650	535	6/74	103.0	71.5	0	81.0	0	0	0	2.8	e
LA SALLE 1 and 2, Seneca, Ill. (Commonwealth Edison Co.)	50-373 50-374	BWR (GE) BWR (GE)	3323 3323	1078 1078	1/84 10/84	102.1 0	103.9 45.9	88.6 97.0	49.8 54.0	0 0	0 0	11.3 0	11.3 15.3	e
LIMERICK 1, Pottstown, Pa. (Philadelphia Electric Co.)	50-352	BWR (GE)	3293	1055	2/86	11.2	0	0	64.7	0	0	0	3.6	e
MAINE YANKEE, Lincoln County, Maine (Maine Yankee Atomic Power Co.)	50-309	PWR (CE)	2560	790	12/72	97.8	72.4	106.5	70.3	5.5	29.9	0	7.7	
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	89.3 99.7	102.0 102.3	22.3 89.6	60.3 72.3	0 0	0 0	77.4 5.6	13.0 9.6	c
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	100.1 98.5 97.2	100.6 10.4 52.5	100.3 0 94.5	70.8 65.7 75.8	0 0 0	0 0 42.0	0 0 0	10.5 14.1 9.8	e
MONTICELLO, Monticello, Minn. (Northern States Power Co.)	50-263	BWR (GE)	1670	545	6/71	98.7	89.1	81.3	71.9	0	0	0	4.1	
NINE MILE POINT 1 and 2, Oswego, N. Y. (Niagara Mohawk Power Corp.)	50-220 50-410	BWR (GE) BWR (GE)	1850 3323	620 1080	12/69 3/88	0 0	0 0	0 0	58.3 26.3	0 0	100.0 0	100.0 0	20.0 21.3	c,e c
NORTH ANNA 1 and 2, Louisa County, Va. (Virginia Electric & Power Co.)	50-338 50-339	PWR (West) PWR (West)	2775 2775	907 907	6/78 12/80	98.0 77.8	68.8 44.0	0 0	61.4 71.8	0 0	12.2 0	100.0 0	14.5 7.7	e

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, %				Footnotes
			MW(t)	MW(e)		Jan.	Feb.	Mar.	Cumulative (lifetime)	Jan.	Feb.	Mar.	Cumulative (lifetime)	
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	4.6 99.4 89.8	41.2 94.6 100.3	87.4 97.2 98.3	66.8 67.8 68.3	44.9 0 9.5	0 4.4 0	11.4 1.7 2.1	12.8 11.0 12.5	e,c
OYSTER CREEK, Oyster Creek, N. J. (Jersey Central Power & Light Co.)	50-219	BWR (GE)	1930	650	12/69	0	0	0	53.7	0	0	84.8	15.3	e,c
PALISADES, Covert Township, Mich. (Consumers Power Co.)	50-255	PWR (CE)	2200	805	12/71	93.7	0	68.9	43.0	4.9	100.0	3.2	35.0	c
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	101.7 101.2 47.8	94.8 53.7 102.5	10.4 44.3 5.8	58.3 71.7 86.4	0 0 49.0	0 45.9 0	85.8 52.5 71.8	27.2 8.7 9.1	c c c
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	0 0	0 0	49.3 52.2	0 0	0 0	0 0	14.6 13.3	m m,c
PERRY 1, Perry, Ohio (Cleveland Electric Illuminating Co.)	50-440	BWR (GE)	3579	1205	11/87	89.4	72.3	0	68.5	3.6	0	0	14.6	
PILGRIM 1, Plymouth, Mass. (Boston Edison Co.)	50-293	BWR (GE)	1998	655	12/72	0	0	10.7	45.7	0	0	29.5	12.5	m,c
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	102.7 103.5	102.1 102.8	88.8 93.3	73.7 80.9	0 0	0 0	0 8.5	2.0 1.2	
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	105.0 104.1	104.9 103.9	104.5 74.3	79.8 84.2	0 0	0 0	0 0	6.3 2.9	 e
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	94.9 94.4	98.1 95.4	95.8 84.8	66.0 63.0	0 0	0 0	0 8.8	5.0 8.3	
RANCHO SECO, Sacramento County, Calif. (Sacramento Municipal Utility District)	50-312	PWR (B&W)	2772	918	4/75	53.3	0	28.3	39.6	31.4	99.8	65.0	43.4	c
RIVER BEND 1, St. Francisville, La. (Gulf States Utilities Co.)	50-458	BWR (GE)	2894	934	6/86	76.9	55.5	32.2	60.0	2.3	20.2	0	10.2	e

(Table continues on the next page.)

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, % Jan.	MDC Feb.	MDC Mar.	MDC Cumulative (lifetime)	Forced outage rate, % Jan.	FOR Feb.	FOR Mar.	FOR Cumulative (lifetime)	Footnotes
ROBINSON 2, Hartsville, S. C. (Carolina Power & Light Co.)	50-261	PWR (West)	2200	700	3/71	0	0.6	97.5	63.9	0	17.1	5.5	14.0	e
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	98.1 58.3	74.3 71.6	71.7 85.0	55.9 52.1	0 38.8	22.6 15.4	15.5 13.9	23.4 30.2	
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	0 36.9 93.4	0 54.6 101.2	0 101.5 102.1	51.1 68.5 64.5	0 62.6 7.8	0 39.0 0	0 0 0	19.5 5.3 8.4	e c
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	96.9 38.0	85.3 0	97.9 0	34.9 38.8	0 0	11.1 0	0 0	55.2 49.4	 e
SHEARON HARRIS 1, Bonsal, N. C. (Carolina Power & Light Co.)	50-400	BWR (GE)	2775	900	5/87	95.2	78.6	95.2	71.8	4.0	10.5	2.7	6.6	
SHOREHAM, Brookhaven, L.I., N. Y. (Long Island Lighting Co.)	50-322	BWR (GE)	2436	819										k
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	45.5	0	84.6	60.6	27.8	0	0	11.7	c k,i
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	101.7 100.9	102.8 0	101.5 0	71.2 82.3	0 0	0 0	0 0	3.6 5.7	 e
SUMMER 1, Broad River, S. C. (South Carolina Electric & Gas Co.)	50-395	PWR (West)	2775	900	1/84	89.4	56.5	11.4	68.3	0	16.4	67.6	7.6	c
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	0 0	0 0	0 0	55.5 58.4	100.0 0	100.0 0	100.0 0	21.0 14.2	c e
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	57.9 99.8	72.1 97.0	89.8 62.8	70.1 76.7	39.7 0	23.9 4.0	2.3 16.8	10.2 7.3	
THREE MILE ISLAND 1, Three Mile Island, Pa. (Metropolitan Edison Co.)	50-289	PWR (B&W)	2772	906	12/78	104.4	104.3	104.2	41.8	0	0	0	52.5	
TROJAN, Columbia, Oreg. (Portland General Electric Co.)	50-344	PWR (West)	3411	1130	5/76	101.2	99.3	100.9	56.5	0	0	0	13.3	

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, % Jan.	Feb.	Mar.	Cumulative (lifetime)	Forced outage rate, % Jan.	Feb.	Mar.	Cumulative (lifetime)	Footnotes
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	63.6 0	91.5 0	62.8 62.3	100.0 0	31.2 0	0 0	13.2 11.4	c e
VERMONT YANKEE, Vernon, Vt. (Vermont Yankee Nuclear Power Corp.)	50-271	BWR (GE)	1593	514	11/72	88.8	29.1	0	71.0	0	0	0	6.0	e
VOGTLE 1 and 2, Waynesboro, Ga. (Georgia Power Co.)	50-424 50-425	PWR (West) PWR (West)	3411 3411	1157 1157	6/87	68.5	82.5	101.0	72.8	27.3	8.7	0	14.5	 k
WASHINGTON NP 2, Richland, Wash. (Washington Public Power Supply System)	50-397	BWR (GE)	3323	1100	12/84	76.2	68.3	73.0	57.9	6.5	6.1	0	9.3	
WATERFORD 3, Taft, La. (Louisiana Power & Light)	50-382	PWR (CE)	3410	1104	9/85	90.3	102.4	100.3	76.4	8.4	0	0	7.5	
WOLF CREEK 1, Burlington, Kans. (Kansas City Power & Light Co.)	50-482	PWR (West)	3411	1170	9/85	83.8	93.2	101.6	71.2	1.4	6.3	0	6.3	
YANKEE ROWE, Rowe, Mass. (Yankee Atomic Electric Co.)	50-29	PWR (West)	600	175	11/60	42.1	99.0	101.5	74.4	0	2.3	0	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	87.0 28.7	16.5 83.4	72.5 98.3	58.3 61.8	7.1 53.0	79.1 8.1	12.9 0	12.7 13.5	c c

[a]The information in this table is derived from NRC Publication NUREG-0020.
[b]License suspended.
[c]Outage for maintenance, repair, or inspection.
[d]Down for modifications.
[e]Outage for refueling.
[f]Operating summary not supplied by utility.
[g]Outage following scram.
[h]Power restriction.
[i]Unit in power ascension (full-power license, not yet in commercial operation).
[k]Fuel loading and low-power testing.
[l]Not available as this issue went to press.
[m]Administrative shutdown.
[n]Other.
[o]Fuel loading and precritical testing.
[p]DER capacity; MDC capacity not available.

Table 3 Power Generation During the First Quarter of 1989

Power generation	1987	1988	January	February	March	Year to date
Gross electrical, MW(e)h	469 212 562	549 213 799	48 748 333	40 813 931	41 757 727	131 652 863
Net electrical, MW(e)h	445 789 016	521 697 302	46 308 267	38 737 380	39 628 461	124 996 460
Average unit factors, %						
Service	67.2	69.8	68.7	65.2	61.2	65.3
Availability	67.2	69.8	68.7	65.2	61.2	65.3
Capacity						
MDC	62.0	65.1	65.3	61.3	56.4	61.1
DER	60.5	63.6	63.7	59.8	54.9	59.6
Forced outage rate	15.1	10.2	11.6	13.4	15.1	13.7

3. Monitor the surge line to assess the extent of TS or to obtain data through collective efforts with other utilities, beginning with hot functional testing;

4. Update their stress and fatigue analyses to ensure compliance with applicable codes within one year after a low power testing license is issued.

NRC ADDS CALVERT CLIFFS, NINE MILE POINT 2, AND SEQUOYAH 1 TO PROBLEM REACTOR LIST

In January 1989 the NRC placed three plants, Calvert Cliffs, Nine Mile Point 2, and Sequoyah 1, to its "Problem Plant" list while removing Dresden and Rancho Seco from that dubious distinction roster.[5] Six others, Nine Mile Point 1, Sequoyah 2, Turkey Point 3 and 4, Fermi 2, and Fort Calhoun, also still remain on the list.

According to the NRC, three recent events at Calvert Cliffs involving failures to use plant procedures properly caused the NRC action with respect to that unit. The first event occurred June 6, 1988, when an employee failed to return a voltage regulator on an emergency diesel generator from a manual position to an automatic position for a 48-hr period. Plant officials said that, because of this mistake, the regulator would not

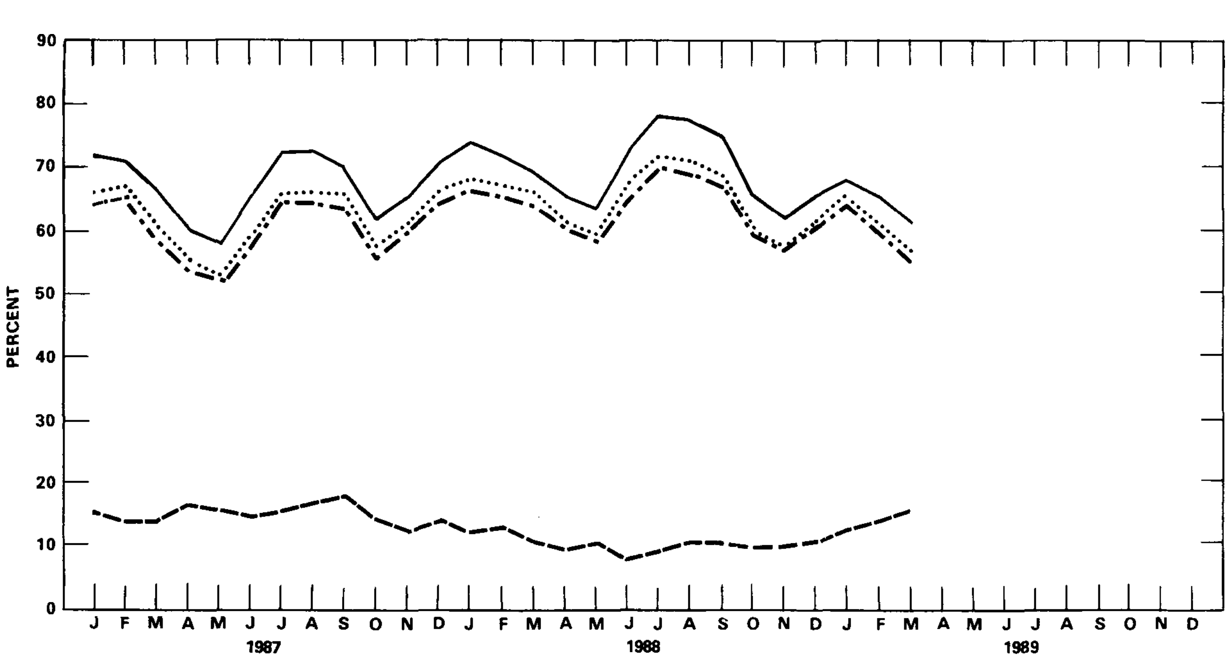

Fig. 1 Average unit availability, capacity factors, and forced outage rates. ———, availability factor. · · · · ·, MDC capacity factor. · — · —, DER capacity factor. — — —, forced outage rate.

have been available for instant startup in an emergency situation.

The second incident happened July 4, 1988, during reactor startup testing when an employee failed to calibrate a reactor power meter properly for 12 hr. The third, and most serious, event took place on Sept. 15, 1988, when an employee drowned in a large storage tank while performing maintenance work.

A spokesman for NRC said that these events follow previously expressed agency concerns that the Baltimore Gas and Electric Company (BG&E) had experienced a decline in performance in engineering support and technical issues. He continued that the Commission recognized BG&E for its efforts taken to improve performance, including increased onsite engineering support and staff reorganization, but added that NRC wants more to be done to effect a turnaround in overall plant performance.

E. A. Crooke, President and Chief Operating Officer of BG&E, expressed regret over these developments saying that the utility recognizes the seriousness of this action. He said that a task force has been created to address NRC's latest concerns and look into plant operations. "Our goal in all of this is to promptly return Calvert Cliffs to a position of excellence in the commercial nuclear industry," he asserted. "We will be satisfied with nothing less."

The NRC said that Nine Mile Point 2 was added to the list mainly because of questions raised concerning the presence of Unit 1 on the list. The agency explained that, since Niagara Mohawk Power Corp. has the same management and training for both units, it placed Unit 2 on the list to verify that there are no problems with the reactor.

The Commission also remarked that it was concerned that the unit experienced more shutdowns, safety-system actuations, and personnel errors in 1988 than was expected. A company spokesman stressed, though, that the plant's problems are not safety related but rather affect efficiency.

He noted that the "tremendous" amount of time, money, effort, and personnel changes expended to correct the problems of Unit 1 will benefit Unit 2. Following the placement of Unit 1 on NRC's list, Niagara Mohawk created an ongoing long-range improvement plan to perpetuate all improvements and lessons learned and a nuclear oversight committee consisting of board members and the executive vice president (nuclear).

The last reactor, Tennessee Valley Authority's Sequoyah 1, was removed from an NRC shutdown list and added to NRC's problem list after it restarted at the end of 1988. The unit returned to service Nov. 6, 1988, after being ordered to close by NRC in 1985 because of safety and management concerns.

NRC ISSUES POLICY STATEMENT ON NUCLEAR POWER-PLANT OPERATIONS

An NRC Policy Statement was issued on Jan. 24, 1989, effective as of that date, which sets forth the Commission's expectations of utility managers and licensed operators at nuclear power plants regarding the conduct of control-room operations. The summary of the document reads as follows:[6]

> SUMMARY: This policy statement is being issued to make clear the Commission's expectation of utility management and licensed operators with respect to the conduct of nuclear power plant operations. The Commission believes that it is essential that utility management at each nuclear power reactor facility establish and maintain a professional working environment with a focus on safety in control rooms and throughout the plant. The Commission also believes that each individual licensed by the NRC to operate the controls of a nuclear power reactor must be keenly aware that he or she holds the special trust and confidence of the American people, conferred through the NRC license, and that his or her first responsibility is to assure that the reactor is in a safe condition at all times. This policy statement specifically describes the Commission's expectations of utility management and licensed operators in fulfilling NRC regulations and prior guidance regarding the conduct of control room operations. The policy statement further provides the Commission's endorsement of industry initiatives to enhance professionalism by both management and plant operators.

The release then goes on to cite the background that led to the issuance of the statement. This background states, in part:[7]

> It is essential that control room operators are (1) well trained and qualified, (2) physically and mentally fit to carry out their duties, and (3) attentive to plant status relevant to their responsibilities to ensure the continued safe operation of nuclear facilities. It is also essential that management at each nuclear power reactor facility establish and maintain a professional working environment in which the licensed operator may be fully successful in discharging his or her safety responsibilities.
>
> On a number of occasions, the NRC has received reports and has found instances of operator inatten-

tiveness and unprofessional behavior in control rooms of some operating facilities. Reported instances include: (1) licensed operators observed to be apparently sleeping while on duty in the control room or otherwise being inattentive to their license obligations, (2) operators using entertainment devices (for example, radios, tape players, and video games) in the control room in a way that might distract their attention from required safety-related duties, and (3) unauthorized individuals being allowed to manipulate reactivity controls. Such conduct is unacceptable and inconsistent with the operators' licensed duties.

The Commission has previously addressed its expectations of operator conduct in Commission regulations and regulatory guidance. Under 10 CFR 50.54(k), "An operator or senior operator licensed pursuant to Part 55 of this chapter shall be present at the controls at all times during the operation of the facility."[1] The continuous presence of a senior operator in the control room to ensure that the operator at the controls is able to perform the actions and/or mitigate an accident is required by §50.54(m)(2)(iii). Commission regulations in 10 CFR Part 55 establish standards for licensing nuclear power plant operators.

The Commission has addressed operator training and qualifications and fitness-for-duty in policy statements. The policy statement on training and qualifications endorsed the Institute of Nuclear Power Operations (INPO)-management Training Accreditation Program. The policy statement on fitness for duty endorsed the concept that the workplace at nuclear power plants is to be drug and alcohol free. Fitness-for-duty rulemaking is under consideration by the Commission.

Guidance regarding the conduct of licensed operators and control room operations has been addressed in an NRC Circular and in NRC Information Notices. Specifically, IE Information Notice 79-20, Revision 1, emphasized that only licensed operators are permitted to manipulate controls [10 CFR 50.54(i)] and that a licensed operator is required to be present at the controls during facility operation [10 CFR 50.54(k)]. IE Circular 81-02 provided the following guidance: (1) knowledge of the plant's status must be ensured during shift changes by a formal watch turnover and relief, (2) licensed operators must be alert and attentive to instruments and controls, (3) potentially distracting activities in the control room must be prohibited, (4) access to the control room must be limited, and (5) eating and training activities should not compromise operator attentiveness or a professional atmosphere. Information Notice 85-53 reiterated the guidance of IE Circular 81-02.

In Information Notice 87-21, the NRC informed all nuclear power reactor facilities and licensed operators about certain licensed operators observed to be apparently sleeping while on duty. The notice reaffirmed the necessity for high standards of control room professionalism and operator attentiveness to ensure safe operation of nuclear power facilities. Further, Information Notice 88-20 reiterated the concern about unauthorized individuals manipulating controls and performing control room activities.

The Commission is aware that the industry has taken action to foster the development of professional codes of conduct by operators and has worked toward establishing management principles for enhancing professionalism of nuclear personnel. The Commission believes that such an operator code of conduct developed by operators and supported by utility management can contribute to operator professionalism and commends the industry and especially the operators who contributed to these efforts. . . .

The Commission has decided to issue this policy statement to help foster the development and maintenance of a safety culture at every facility licensed by the NRC, and to make clear its expectations of utility management and licensed operators in fulfilling NRC regulations and prior guidance regarding the conduct of control room operations.

The policy statement itself is as follows:[8]

Policy Statement

The Commission believes that the working environment provided for the conduct of operations at nuclear power facilities has a direct relationship to safety. Management has a duty and obligation to foster the development of a "safety culture" at each facility and to provide a professional working environment, in the control room and throughout the facility, that assures safe operations. Management must provide the leadership that nurtures and perpetuates the safety culture. In this context, the term "safety culture" is defined as follows:

The phrase "safety culture" refers to a very general matter, the personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants. The starting point for the necessary full attention to safety matters is with the senior management of all organizations concerned. Policies are established and implemented which ensure correct practices, with the recognition that their importance lies not just in the practices themselves but also in the environment of safety consciousness which they create. Clear lines of responsibility and communication are established; sound procedures are developed; strict adherence to these procedures is demanded; internal reviews are performed of safety related activities; above all, staff training and education emphasize the reasons behind the safety practices established, together with the consequences for safety of shortfalls in personal performance.

These matters are especially important for operating organizations and the staff directly engaged in plant operation. For the latter, at all levels, training emphasizes the significance of their individual tasks from the standpoint of basic understanding and knowledge of the plant and the equipment at their command, with special emphasis on the reasons underlying safety limits and the safety consequences of violations. Open attitudes are required in such staff to ensure that information relevant to plant safety is freely communicated; when errors of practice are committed, their admission is particularly encouraged. By these means, an all pervading safety thinking is achieved, allowing an inherently questioning attitude, the prevention of complacency, a commitment to excellence, and the fostering of both personal accountability and corporate self-regulation in safety matters.

Nuclear power plant operators have a professional responsibility to ensure that the facility is operated safely and within the requirements of the facility's license, including its technical specifications and the regulations and orders of the NRC. Mechanical and electrical systems and components required for safety can and do fail. However, the automated safety features of the plant, together with the operator, can identify at an early stage degradation in plant systems that could affect reactor safety. The operator can take action to mitigate the situation. Therefore, nuclear power plant operators on each shift must have knowledge of those aspects of plant status relevant to their responsibilities, maintain their working environment free of distractions, and using all their senses, be alert to prevent or mitigate any operational problems. Each individual licensed by the NRC to operate the controls of a nuclear power reactor must be keenly aware that he or she holds the special trust and confidence of the American people, conferred through the NRC license, and that his or her first responsibility is to assure that the reactor is in a safe condition at all times.

The following criteria reflect the Commission's expectations concerning the conduct of operations in control rooms and licensed operators at nuclear reactors consistent with 10 CFR 50.54 and guidance provided in an NRC Circular and Information Notices:

— Conduct within the control room should always be professional and proper, reflecting a safety-minded approach to routine operations. The operator "at the controls" and the immediate supervisor must never relinquish their safety responsibilities unless properly relieved, including a thorough turnover briefing, by a qualified operator.

— Activities within the control room should be performed with formality. Operator actions must be in accordance with approved procedures. Verbal communications should be clear and concise. Appropriate consideration should be given to the need for acknowledgement and verification of instructions received.

— The control room of a nuclear power plant, and in particular the area "at the controls," must be secure from intrusion. Access should be strictly controlled by a designated authority; only authorized personnel should be permitted to be present in the control room; and regulatory restrictions concerning manipulation of the controls must be meticulously observed.

— The operator at the controls, and the immediate supervisor, must be continuously alert to plant conditions and ongoing activities affecting plant operations, including conditions external to the plant such as grid stability, meteorological conditions, and change in support equipment status; operational occurrences should be anticipated; alarms and off-normal conditions should be promptly responded to; and problems affecting reactor operations should be corrected in a timely fashion.

— Activities within the control room should be limited to those necessary for the safe operation of the plant. Management should provide the direction, facilities, and resources needed to accommodate activities not directly related to plant operations.

— Activities outside the control room with the potential to affect plant operations, such as on-line maintenance and surveillance, should be fully coordinated with the control room. Effective methods for communication with or notification of the operator at the controls should be established and maintained throughout each evolution.

— Written records of plant operations must be carefully prepared and maintained in accordance with requirements for such records and in sufficient detail to provide a full understanding of operationally significant matters.

— The working environment in the control room should be maintained to minimize distractions to the operators. Management should act to remove distractions that would interfere with the operator's ability to monitor the plant either audibly or visually, including work activities that are not related to the operator's immediate responsibility for safe plant operation. Consideration should be given to reducing environmental distractions such as lighted alarms that are not operationally significant, or alarms that signify normal operating conditions.

— Foreign objects and materials not necessary for plant operations, ongoing maintenance, or surveillance testing should be restricted from the area "at the controls" to preclude inadvertent actuation of the controls or contamination of control devices.

NRC STAFF COMMENTS ON MARK I CONTAINMENT SAFETY

At a public meeting on Jan. 26, 1989, the NRC staff presented their recommendations regarding the safety of the Mark I containment [a containment design on certain General Electric (GE)-built boiling-water reactors (BWRs)] to the Commission.[9] Their principal conclusion was that, despite the relatively small volume inside the Mark I containments, plants equipped with such containments do not pose a greater risk of containment failure than do other types of plants. However, the staff also concluded that a Mark I containment vessel could be "challenged" in the event of a large-scale core-melt accident.

Staff members stressed that it was not urgent for NRC to take any action regarding the nation's 24 Mark I plants. They did recommend, however, that certain modifications be made to improve their performance during the unlikely event of a severe core melt. The staff suggested that modifications be made on a plant-by-plant basis parallel to the ongoing individual plant evaluation program.

In response to a question from Commissioner K. M. Carr, the staff said it would not be necessary to shut down the plants to begin the modifications but that the implementation of the changes could be made during a scheduled refueling outage.

The briefing outlined three Mark I failure modes with positive risk importance. They are: (1) overpressurization leading to core damage (i.e., containment failure before core melting), (2) overpressurization corium–concrete interaction plus steam, and (3) overtemperature corium–concrete interaction.

According to the staff, containment venting is particularly important in reducing overall risk. The staff said they disagree with the nuclear industry's assessment that venting procedures are adequate. Among other concerns, staff members concluded that plant operators would be reluctant to vent steam during an emergency because the reactor building's sheet metal ductwork would be likely to fail and thus endanger repair crews.

Altogether, the staff advanced five recommendations for remedial actions: (1) accelerate NRC actions to implement the station blackout rule, (2) require alternate water supply for drywell spray/vessel injection with pumping capability independent of normal and emergency air conditioning (AC), (3) require hardened venting capability from wetwell and require isolation valves to be remotely operable independent of normal and emergency AC, (4) require enhanced automatic depressurization system reliability and additional power and/or nitrogen supply and cable reliability, and (5) require implementation of improved emergency procedure guidelines.

The staff asserted that, taken together, the entire package of improvements will lower core-melt probabilities and reduce the potential for containment failure. In addition, the staff maintained that these modifications are cost effective when considering man-rem averted relative to installation costs. They admitted, however, that the nuclear industry generally does not share this belief. At Boston Edison's Pilgrim 1, the only plant to implement all the suggested modifications, the estimated cost was $5.6 million.

After the conclusion of the presentation, Commissioner T. M. Roberts said that in his view the staff was using a "buckshot" approach to safety. The Commissioner wondered why this issue, as well as others, such as accident management, could not all be addressed in the individual plant evaluations (IPEs). The staff responded by saying that they felt the Mark I changes should be in addition to, rather than part of, the IPEs.

PROGRESS TOWARD RESTART OF PEACH BOTTOM

As previously reported here,[10] both Peach Bottom reactors have been shut down since March 1987 as the result of control-room personnel deficiencies that included sleeping on the job by reactor operators. Ever since, Philadelphia Electric Company (PECO) has been working to upgrade its management system to satisfy NRC requirements. PECO hoped to achieve restart by April 1989.

On Feb. 6, 1989, the NRC Commissioners met with PECO executives to review the progress made in improving Peach Bottom management.[11]

In his opening remarks, PECO Chairman J. F. Paquette, Jr., noted that in a previous meeting held on Oct. 5, 1988, the Commissioners were told of substantial managerial changes made at Peach Bottom in an effort to win restart approval. Paquette also said the company has tried to create a "safety culture" at the plant and added that an

NRC inspection had begun that day to assess the extent and effectiveness of the changes.

Dickinson M. Smith, PECO vice president for Peach Bottom, told the Commission that the decision-making organization at Peach Bottom had been consolidated into one vice president and that three new managerial positions had been created to ease the workload on the plant manager. According to Smith, operator readiness for restart had been enhanced through the creation of six rotating shifts, each with three senior reactor operators and two reactor operators. PECO also made extensive use of reactor simulators to prepare its staff for restart.

In the area of security, Smith admitted that, although Peach Bottom had previously had a poor security program, PECO now feels that it has substantially improved security through the hiring of a new contractor with increased PECO oversight, overtime controls for guards, and improved facilities and equipment. Smith mentioned that security at Peach Bottom had been the focus of an NRC inspection in late January 1989.

Physical plant modifications and maintenance were also covered in Smith's presentation. The breakdown given included approximately 20 modifications to be installed and 400 work orders to be processed before Peach Bottom can restart. Compared to one year ago, when there were 2900 items of overdue preventive maintenance, there are now zero items, Smith reported.

Smith was questioned by NRC Commissioner Carr regarding Smith's assurance to the Commissioners that by restart the plant would have reached the normal industry backlog of 1000 corrective maintenance work orders. Carr asked why the figure could not be brought down even lower; Smith responded that below a certain point it would not be cost effective to try to decrease the backlog further.

Regarding the health aspects of restart, Smith said that more than 90% of the plant had been decontaminated, compound to an industry average of about 18% contaminated. He also indicated that Peach Bottom was below its personnel exposure goal for 1988 and well below its radwaste produced goal. NRC Chairman Zech said that he considers these to be "significant" indicators of performance and that he was "pleased" by PECO's efforts.

The next presentation was given by C. A. McNeil, Jr., executive vice president for PECO's nuclear group. McNeil told the Commissioners that PECO had five criteria for the restart of Peach Bottom: (1) plant readiness, (2) effective management and staff in place, (3) restart programs implemented, (4) self-assessment capability established, and (5) major technical issues resolved.

As he addressed each of these criteria individually, McNeil reported that, with the exception of some unresolved technical problems, the plant and its personnel were ready for restart. McNeil put special emphasis on PECO's new self-assessment capability which he said was cited as a major weakness when Peach Bottom was closed.

The unresolved technical issues still facing Peach Bottom include: Appendix R (high impedance fault, electrical design review), degraded grid, high-energy line break seals, alternate rod insertion modification, emergency cooling tower tests, and small bore pipe stress.

McNeil said that, because NRC had denied an exemption for the high impedance fault, PECO was now working with NRC staff on a way to resolve the issue. The degraded grid problem was to be addressed through modifications made before restart, and the line break seals were to be corrected by mid-March. According to McNeil, the alternate rod insertion modification was ready for testing and the cooling tower venting problems were to be taken care of by the end of February 1989. McNeil went on to say that small bore pipe stress was common in older plants and should not impede the restart schedule.

The final moments of McNeil's presentation centered on assuring the Commissioners that no major obstacles stood in the way of restart. McNeil indicated that a final agreement now being drawn up with the Commonwealth of Pennsylvania would remove the state from proceedings against PECO in the Third Circuit Court of Appeals and the Atomic Safety and Licensing Board. Refueling at Limerick Unit 1 and low-power operation at Limerick Unit 2 would not hamper the restart of Peach Bottom, McNeil said. He noted that the managements of Limerick and Peach Bottom were completely independent of each other and that there would consequently be sufficient resources for both projects.

Another step toward restart of the Peach Bottom units was taken at the beginning of March 1989 when the Commonwealth of Pennsylvania and PECO reached an agreement whereby, in return for a number of assurances by PECO, the Commonwealth would drop its actions against PECO in the Third U.S. Circuit Court of Appeals and before the NRC's Safety and Licensing Board.[12]

The agreement, which requires approval by the NRC, will give the Commonwealth of Pennsylvania unusual access to confidential plant records and will allow state officials to monitor the operation of the plant closely. As indicated by PECO executive vice president McNeil, the specter of a prolonged court battle was a strong incentive to reach an accord. Corbin said, "We have to work [with the state] rather than face them in court."

The accord allows Pennsylvania officials to view evaluation reports prepared on Peach Bottom by the Institute of Nuclear Power Operations (INPO) for 1987 to 1992. The INPO is a utility group that evaluates the performance of its members. State Environmental Resources Secretary A. A. Davis said the Commonwealth would be the first state in the nation to gain access to INPO evaluations.

Other documents covered in the accord include overtime records for control-room personnel, PECO internal interviews with reactor operators, and inspection reports by the utility's insurers. In addition to document privileges, PECO agreed to allow state inspectors unescorted access to the plant.

Other concessions, and ones that will become part of Peach Bottom's license, include restrictions on control-room overtime, termination of contractors who discipline "whistle blowers," the creation of programs to spot potential safety problems, and restrictions on transfers of Peach Bottom employees to the Limerick Generating Station. These commitments will be enforced by NRC.

On February 21, INPO notified PECO that the company's corrective actions at Peach Bottom have been "responsive and effective" and complimented the utility for implementing "significant change" both in the company and at the plant. An earlier letter by INPO, dated Jan. 11, 1988, had been very critical of PECO. That letter had noted several deficiencies at Peach Bottom and called for "fundamental change in the corporate approach to nuclear operations." The most recent INPO communication concludes that "PECO has adequately addressed the recommendations contained in that letter and. . . that actions taken. . . to resolve additional issues raised in subsequent INPO visits have been responsive and effective."

The Advisory Committee on Reactor Safeguards (ACRS) also reviewed the Peach Bottom restart effort and reported on its conclusions in a letter on Mar. 14, 1989 (Ref. 13). The letter reads, in part, as follows:

> During the 347th meeting of the Advisory Committee on Reactor Safeguards, March 9–11, 1989, we reviewed the Philadelphia Electric Company's (licensee's) plans for restart of the Peach Bottom Atomic Power Station and the evaluation of these plans by the NRC staff. Our Subcommittee on General Electric Reactor Plants, which considered the Peach Bottom Restart, met with representatives of the licensee and the NRC staff on March 9, 1989 to discuss this matter. . . .
>
> Since the transmittal of the shutdown order to the licensee on March 31, 1987, there have been major changes in corporate and plant management, in staffing, training, and procedures. There has also been a significant and largely successful effort to eliminate overdue preventive maintenance items.
>
> The NRC staff is prepared to conclude that, subject to completion of certain well-defined commitments to modifications of equipment and revisions of procedures, the licensee can, with the organization now in place, operate the Peach Bottom Atomic Power Station without undue risk to the health and safety of the public. We find no reason to disagree with the staff's position.
>
> Since the staff's evaluations have been made for a managerial and operational team that has not yet operated the plant at power, we endorse the staff's plans to continue a close monitoring and evaluation of this team for an appropriate period after operation at power has begun.

SHOREHAM APPROACHES FULL-POWER LICENSE BUT DISMANTLEMENT APPEARS EVER MORE PROBABLE

A series of separate actions in regard to the embattled Shoreham nuclear power plant seemed, at one and the same time, to bring the facility closer to obtaining a full-power license while apparently inexorably moving in the direction of abandonment and eventual dismantlement of the completed facility. By now the situation is so complicated with so many aspects and entrenched positions that making sense of the entire situation is difficult, at best.

On Feb. 11, 1989, Federal Judge J. B. Weinstein overturned a jury verdict which found the Long Island Lighting Company (LILCO) guilty under the "Racketeering Influenced and Corrupt Organizations" (RICO) Act of having lied to State Regulators about construction progress on Shoreham.[14] In his reversal Judge Weinstein criticized the use of the RICO Act in a civil suit, asserting that applying RICO to such state-regulated industries clearly violates federalism.

The overturning appeared to save LILCO a possible $4 billion in damages unless an appeal did again overturn the judge's decision, but LILCO appeared to have overcome this uncertainty when an agreement was reached between LILCO and the court-appointed attorney for the LILCO ratepayers, J. P. Vladeck. Under that agreement, Vladeck agreed not to appeal the Weinstein decision in exchange for LILCO's agreement to subtract $390 million from its proposed rate increases over the next decade, to pay $10 million for fees for lawyers representing the customers, set up a $10 million fund to satisfy claims of former customers, and consider advice from a Citizens' Advisory Panel made up of consumer groups.

This appeared to settle the matter and thus pave the way for LILCO to go through with last year's plan by Governor Cuomo to sell Shoreham for $1 to the New York State agency that would dismantle it, except that the Suffolk County legislature voted unanimously not to drop the RICO suit against LILCO. They said they would not consider dropping the suit unless LILCO promises to "irrevocably withdraw its request for an operating license for Shoreham" and pay the county $23 million in damages and lawyer's fees. Everyone else, including (in a rare accord) LILCO and Governor Cuomo, deplored the county's stance, which was widely seen as an attempt to profit from LILCO's financial distress.

In a meeting held February 15, lawyers for Suffolk County, Judge Weinstein, and mediator Feinberg discussed ways to reach a settlement. Although Feinberg portrays the meeting as "constructive," others present say Weinstein summarily rejected Suffolk County's demand that LILCO "irrevocably close" Shoreham in return for a promise not to appeal.

Also on February 15th, the New York Public Service Commission (NYPSC) unanimously approved a temporary 5.4% rate increase for LILCO. The increase is good only for 90 days, at which point the NYPSC was to decide if it should be made permanent. If the increase were kept in place for the next year, it would bring in approximately $97 million for the utility. The increase, however, was conditional on LILCO not operating Shoreham while negotiations over its future are in progress. The vote to tie the increase to the Shoreham provision was split 4 to 3.

After the NYPSC decision, its Chairman, P. A. Bradford, said he was not concerned about the effect the increase would have on the continuing negotiations. "We have to call them as we see them," he said, "and these rates were justified." Wall Street experts say the ruling bolstered LILCO's position by showing regulatory support for the utility.

A LILCO spokesperson said that the utility welcomed the increase but was concerned about the restrictions placed on Shoreham. The company was studying the order to see if an appeal is possible, she reported.

Even if Suffolk County and LILCO were to agree on an arrangement to close Shoreham, the deal would have to be approved by Cuomo and possibly the state legislature. Cuomo arranged such a deal in December, only to see it die for lack of legislative approval. Cuomo's spokesman says the governor's position was that several parts of any potential agreement would need the legislature's approval. He also said that Cuomo would want to be sure a settlement would not be overridden by the legislators.

Late this week the governor said the legislature had erred in rejecting his proposal and if current events continued, there would be "an open plant and even higher electric rates on Long Island." State Sen. J. L. Lack (R-Hauppauge), however, reiterated that the legislature does not have, or want, any role in setting rates.

Meanwhile, LILCO signed six contracts to purchase 400 MW of power from Long Lake Energy Corp., a developer of third-party cogeneration projects. LILCO will purchase, for 20 yr, the surplus electricity of the gas-fired projects, which are supposed to go on line in the early 1990s.

Then, in early March 1989 Governor Cuomo and LILCO reached another agreement to close Shoreham, this time without involving the New York State legislature.[15]

Under the terms of the settlement, the state would buy Shoreham for $1 and then decommission and dismantle it. The two sides also agreed to withdraw from several administrative and court proceedings concerning Shoreham and other issues. Unlike a deal brokered by Cuomo in June 1988, the document makes no mention of increased electric rates for LILCO. Instead, new rates would be set by the NYPSC. A LILCO statement said the company expects the rate increases to be "similar" to the 5% a year for 10 yr proposed in the June agreement. LILCO also said the company's board of directors would not vote on the plan until a rate schedule is approved by the Commission.

In his statement, Cuomo predicted that rates under this agreement would be even lower than under the original proposal. He based this assessment on "additional rate mitigation measures, including a gross receipts tax reduction" bill he plans to submit to the legislature. As part of the accord, the parties also agreed to support an allocation to LILCO of a minimum of $100 million per year for 5 yr from New York State industrial development bonds. A spokesman for Cuomo said the funds would be used to offset the impact of the settlement on ratepayers.

In addition to the board of directors, the accord would have to be approved by the company's shareholders, the Long Island Power Authority, the New York State Power Authority, and the NYPSC. Each of these bodies supported Cuomo's previous plan.

It appeared that only the New York State Power Authority could pose a potential problem for the new proposal since the authority's present members do not support closing Shoreham. Cuomo, however, has nominated new trustees who do support his position, and if they are confirmed as expected, the authority would almost certainly back the plan.

In Albany, members of the state legislature mostly expressed approval of the agreement and relief that the issue finally seemed settled. Assembly Speaker M. Miller (D-Brooklyn) said he agreed with the Governor that "rate-making decisions should be made by the Public Service Commission." Even members who were less than enthusiastic about the settlement agreed that it was unlikely it could be changed now.

Assemblyman P. Harenberg (D-Bayville) called the agreement "very burdensome on the ratepayers" and said it "will have a devastating impact on the Long Island economy." He also said, however, that there was little the legislature can do in the face of a "fait accompli."

Others who were dissatisfied with the arrangement included the Long Island Association, the area's largest business group. Its Chairman, J. V. N. Klein, said "if it is the same proposal [as that of 1988] with a different suit on, we will oppose it."

The greatest challenge, however, could come from Suffolk County. Since the county refused to accept Judge Weinstein's dismissal of the fraud verdict against LILCO, it could very well challenge any rate increases granted by the PSC under the plan.

However, LILCO's acceptance of the plan to close Shoreham came even as the utility continued its efforts to get the plant licensed. Although NRC officials denied it, rumors had circulated that the Commission was preparing to license the plant in the near future. Indeed, Cuomo said, in a prepared statement, that "with NRC about to act, the clock is running out on our ability to prevent Shoreham from firing up." LILCO said it would continue to "vigorously" pursue a full-power license for Shoreham. The new agreement with Governor Cuomo, however, required that the plant not be operated before April 15, which would give state agencies a chance to vote on the plan. If those agencies approve the deal, LILCO would not operate the plant at all unless its shareholders veto the agreement; the utility expected to have a shareholders vote by June 15.

Throughout the Shoreham odyssey, LILCO and the state agreed that Long Island would face a severe power shortage without the 809 MW of electricity provided by the nuclear plant. Under the agreement with the Governor, LILCO can request that the New York Power Authority build as many as three new oil- or natural gas-fired generating facilities. In the short-term, however, energy industry experts are concerned that brownouts and blackouts could occur until 1991 when an underwater transmission cable will give Long Island access to cheaper power from upstate New York and Canada. The cable would be able to carry up to 600 MW of power.

Meanwhile, quite as though LILCO were really trying to operate Shoreham, the NRC moved closer to granting a full-power license when, on Mar. 6, 1989, the Commissioners voted 4 to 0 to dismiss officials from New York, Suffolk County, and the town of Southhampton from NRC proceedings involving Shoreham. The Commissioners said the officials were guilty of "bad faith" and a "willful attempt to obstruct the Commission's proceedings." An NRC spokesman said that with this obstacle gone, a full-power license for the plant could be considered in "a month or so."

Late in March 1989 Judge Weinstein dismissed objections to the settlement of the RICO case brought by Suffolk County; he called the settlement "well within the range of reasonableness."[16]

On still another front of efforts to give Shoreham a full-power license, an NRC operation readiness inspection team gave Shoreham "high marks" in anticipation of issuance of a commercial license. W. Russell, NRC's region I administrator, said the inspection only turned up six items that needed to be corrected before a license could be issued. Russell indicated the plant could be ready to operate as early as April 1989.

The six items that still needed to be addressed were: (1) a backlog of work orders, (2) the reworking of certain equipment on which maintenance has been done more than once, (3) planning for an orderly transition from contractor-based plant operation to the greater use of utility employees, (4) correction of an intermingling of Class I and II fasteners, (5) pressure transmitters that have electrical cables that are sealed in places different from what was stated in the environmental qualifications approval, and (6) limit switches that are also installed differently than what was approved in the environmental qualifications.

FIFTEEN NEW FINES DURING REPORTING PERIOD

Fifteen civil penalty fines have been levied by the NRC on reactor licensees during the three-month period covered by this report (January, February, and March 1989), and one fine imposed earlier was reduced. In each case the affected utility must report to the NRC on the causes and proposed corrections of the problem that led to the fine and has 30 days from its notification to either pay the penalty or protest its imposition in whole or in part. Each of the cases is briefly described.

Quality Assurance Violations at Brunswick

Carolina Power and Light Company (CP&L) had a fine of $75 000 proposed against it in connection with a violation of NRC requirements relating to quality assurance.[17]

During inspections conducted between November and August 1988, the NRC staff identified two instances where the utility failed to promptly identify and take appropriate actions to correct hardware and/or equipment deficiencies at the Brunswick facility.

In one case bolt failures in safety-related motor control centers were first discovered in November 1986, but the initial licensee review failed to identify and correct the root cause of the bolt failures until additional failures were detected in January–February 1988. As a result, associated safety-related equipment might not have operated properly had there been an earthquake.

In the second case an instrument in part of the emergency core cooling system (ECCS) was improperly set. In this instance the Brunswick staff first suspected that a problem might exist in November 1987; however, the deficiency was not confirmed and effective corrective action taken until September 1988. As a result, the high-pressure coolant injection component ECCS might not have been properly protected in the event it experienced a steam line break.

Together these two problems have been categorized as a Severity Level III violation because they reflect the lack of the necessary aggressiveness to ensure that such issues are promptly addressed and resolved as well as a lack of effective communication between various levels of Carolina Power and Light staff at Brunswick.

The base civil penalty for a Severity Level III violation is $50 000; however, in this case it was increased by 50% because of the utility's past history of poor performance in regard to taking prompt and effective action to correct problems.

Security Violations at Trojan

A fine of $75 000 has been proposed against Portland General Electric Company (PGE) in connection with alleged security requirements violations at Trojan.[18]

During an inspection conducted Oct. 17–20, 1988, NRC inspectors found that PGE:

1. Failed to take adequate measures to compensate for the degradation of a security barrier.
2. Failed to record a visitor's entrances and exits from the area.

3. Failed to properly search and escort the visitor.

4. Failed to establish a second security barrier for certain vital equipment.

Collectively, these alleged violations have been categorized as a Severity Level III problem, with Level I being the most serious and Level V the least serious.

The base civil penalty for Severity Level III is $50 000. This penalty was increased to $75 000 because of the company's failure to take thorough corrective actions in response to previous problems with physical barrier degradations identified in January 1987 and July 1988.

Incorrectly Installed Valve Switch at Fermi; Reduction of Earlier Fine

A $50 000 fine was levied against Detroit Edison Company for safety requirements violations involving motor-operated valves (MOVs) at the Fermi plant.[19]

The licensee reported to the NRC that a discharge valve on a recirculation pump failed to close during required tests on Aug. 20 and Aug. 28, 1988. The purpose of the test program was to identify such problems.

Follow-up inspections by NRC inspectors Sept. 6–21 and Oct. 4–6, 1988, determined that the torque switch for the valve that controls the closing force of the valve had been incorrectly installed and calibrated in May 1988. If the discharge valve on the recirculation pump failed to close during a pipe-break accident, it could affect one of the plant's ECCSs, though other ECCSs would not be affected.

As a result of the initial valve failure, the licensee inspected 148 other safety-related MOVs. Deficiencies involving improper installation were found in four of the valve torque switches. All deficiencies were subsequently corrected.

The licensee was cited for four violations of NRC regulations, including the failure by the utility to take prompt corrective action in 1987 and 1988 when it determined that there were programmatic deficiencies in the MOV torque switch installation and calibration program. (Failure to take prompt corrective action resulted in the subsequent failure of the recirculation pump MOV in August.)

Other violations included: (1) failure to maintain and control the proper range for torque switch settings, (2) failure to require contract personnel to have the necessary skills to perform maintenance on the MOVs, and (3) failure to provide adequate procedures for some valve tests.

The NRC, however, has also reduced a fine proposed earlier against Detroit Edison by $25 000. The $200 000 fine was first announced on May 22, 1988, and involved two violations of NRC safety requirements. One of the violations involved a design problem with the reactor containment radiation monitoring system. The NRC staff increased the base civil penalty for this violation by $25 000 because of the licensee's past poor performance in this area. After reviewing the licensee's response to the proposed fine, the NRC concluded that the additional $25 000 for prior performance was unwarranted.

Inadequate Control of Radioactive Liquid Effluents at Rancho Seco

A $100 000 fine against the Sacramento Municipal Utility District (SMUD) resulted from alleged failures by SMUD to control the discharge of radioactive liquids from Rancho Seco.[20]

The violations in question have been the subject of extensive federal review since mid-1986 and were considered by NRC to demonstrate past poor performance, rather than current performance, at Rancho Seco. The NRC recognized that the effect of the radioactive releases was not significant.

During an inspection conducted Apr. 1 to May 23, 1986, NRC inspectors found that SMUD:

1. As of Apr. 1, 1986, failed to establish an adequate surveillance program to provide data on quantities of radioactive material released in liquid effluents.

2. From January 1983 to May 6, 1986, implemented procedures and temporary modifications in the processing and release to the environment of radioactive water without the required Technical Specification and safety review and without required documentation.

3. On June 4, June 6, and June 17, 1985, identified measurable concentrations of the radioactive nuclide Cs-137 in samples of radioactive liquid wastes to be discharged but did not report these concentrations as required.

4. During calendar year 1985 released radioactive materials in liquid effluents such that a member of the public could have received a total body radiation dose of approximately 3.9 mrem, which would exceed the 3 mrem regulatory limit.

5. From Mar. 30, 1983, to Jan. 6, 1986, and from Mar. 6 to Mar. 30, 1986, failed to implement required procedures and to maintain necessary control of liquid radioactive waste handling and discharge to the environment.

6. From Jan. 6 to Mar. 6, 1986, implemented a temporary procedure change for handling radioactive liquids onsite without required approval of the Plant Review Committee.

Collectively, these alleged violations have been categorized as a Severity Level III problem.

The base civil penalty for Severity Level III is $50 000. This penalty was increased to $100 000 because of the duration of the violations and the significant breakdown in management control of the radioactive effluent discharge program. During review, the NRC staff considered all the information provided by SMUD on these issues, their relative age, and the comprehensive corrective actions taken by SMUD prior to NRC authorization of the restart of Rancho Seco on Mar. 30, 1988.

Safety Limit Violation and Records Destruction at Oyster Creek

The NRC staff proposed a $50 000 fine against GPU Nuclear Corporation (GPUN) as the result of an operating license safety limit, numerous violations of standard maintenance procedures, and the alleged destruction, by the former reactor operator who caused the safety limit violation,[21] of a control-room record that provided a chronology of the event.

Both an NRC inspection and an investigation done by the company found that the safety limit violation occurred when the operator mistakenly turned off the fourth of five loops in a reactor water recirculation system while the plant was shut down. At the time, three of the five loops had already been closed, and thus only one such loop was left open. This condition constituted a violation of the NRC requirement that at least two of the five loops in this system be fully open at all times. The violation lasted approximately 2 min, from 2:17 a.m. to 2:19 a.m., on Sept. 11, 1987.

The NRC, as well as a separate company investigation, also found that the operator, after correcting his error by opening two more valves, destroyed a paper tape that provided a chronology of the event. He tore off a portion of the printout that logs control-room alarms, discarded part of it in a trash can, and flushed part of it down a toilet. GPUN subsequently fired him.

In addition, the NRC inspectors found that GPUN failed to follow its maintenance procedures for performing repairs on the piping that feeds cooling water to essential components within the reactor's containment system.

In a letter notifying GPUN of the proposed fine, W. T. Russell, Regional Administrator of the NRC's Region 1, urged company management to focus on the "underlying NRC concern, namely, inadequate control of maintenance and operational activities." In a separate letter to the reactor operator, Mr. Russell pointed out that destroying records was a "serious" violation of his operator's license (which GPUN withdrew in April 1988) and "constitutes an act. . .which cannot be tolerated by the NRC."

Hydrogen Purging System Inoperability at McGuire

A fine of $37 500 was levied against Duke Power Company for alleged violations involving a system for purging hydrogen from areas in the containment building.[22]

The action follows NRC inspections at McGuire which included a review of circumstances associated with the past and current operability of hydrogen skimmer systems at both units. The hydrogen skimmer system is a ventilation system designed to remove hydrogen gas from areas inside the reactor containment building to prevent a dangerous buildup following a postulated major accident.

The NRC staff said the company failed to assure adequate compartment ventilation flow rates necessary to achieve required system performance because a system flow balance was not performed for each train in each unit during their original preoperational testing and no administrative controls had been established to ensure correct compartment damper position since initial startup.

Another violation arose out of a failure by the company to perform an adequate safety evaluation when Unit 2 was restarted in July of 1988 with the hydrogen skimmer system unable to meet design requirements. The NRC said a company evaluation that the unit's hydrogen mitigation system—which uses igniters to burn off excess hydrogen—in combination with a degraded hydrogen skimmer system, would adequately prevent accumulation of hydrogen above allowable levels, was flawed. Plant Technical Specifications do not allow the skimmer system to be inoperable while the plant is running, based upon the operability of the igniter system.

The base civil penalty for these violations is \$50 000, but the fine was reduced by 25% in this case because of the company's comprehensive correction actions.

Failure to Declare Essential Service Water System Inoperable at Wolf Creek

A fine of \$50 000 was proposed against Wolf Creek Nuclear Operating Corporation (WCNOC) in connection with an operating condition problem between Feb. 13 and July 1, 1987 (Ref. 23).

This action resulted from an inspection finding that the plant operated during the specified period in 1987 without declaring inoperable one of its two essential service water systems. Wolf Creek Technical Specifications require that, if one of these two redundant systems is not operable, the plant must be shut down if the inoperable system is not restored within 72 hr. Both essential service water system loops supply cooling water to safety equipment, such as pumps, that would be needed in case of an accident.

One loop was inoperable, NRC contends, because a section of pipe had eroded in some locations to less than 40% of the minimal allowable pipe wall thickness. NRC inspectors first identified this matter as an unresolved issue in 1987. The report of a team inspection conducted June 6–17, 1988, listed it as a potential enforcement item. NRC and WCNOC officials discussed it during an enforcement conference held Nov. 16, 1988, at the NRC regional office in Arlington, Tex.

In his letter informing WCNOC of the enforcement action, R. D. Martin, NRC Regional Administrator, said NRC is concerned that Wolf Creek plant management did not adequately analyze the essential service water system's operability once questions were raised about the erosion problem in early 1987. Mr. Martin said the root cause of the violation appears to have been inadequate involvement in operational matters at the time by the plant's engineering groups.

He pointed out, however, that, since these violations occurred, WCNOC has developed a program to monitor for erosion and corrosion of safety-related piping. In addition, he said, the company has changed procedures to give the engineering staff a defined role in determining the operability of degraded safety systems.

Inadequate Compensation for Degraded Security Boundary at Turkey Point

A fine of \$100 000 was levied on Florida Power and Light Company (FP&L) for alleged security requirements violations at Turkey Point.[24]

The NRC officials notified FP&L on Feb. 1, 1989, that the penalty was being proposed because of the company's failure to provide adequate compensatory measures for a degraded vital area barrier. The NRC said the company repositioned a guard providing compensatory measures for a degraded vital area barrier, which resulted in a failure to control access to vital equipment until identified by the NRC's Senior Resident Inspector 2 days later.

The NRC said the base civil penalty for this violation was \$50 000 but that the amount in this case was doubled as a result of the company's "continued poor performance in security."

The NRC officials said previous civil penalties indicated continued poor performance in the area of security at Turkey Point. They also informed the company in a letter notifying them of the alleged violation and proposed civil penalty that "it is imperative that Florida Power & Light Company management take the necessary action to assure that this pattern of security violations is terminated and to assure that the security of the Turkey Point facility can be adequately maintained."

Missing Secure-Area Keys and Other Security Inadequacies at Maine Yankee

A \$75 000 fine resulted when NRC staff determined that Maine Yankee Atomic Power Company was in violation of NRC security requirements at Maine Yankee.[25]

This fine stems from two NRC inspections of Maine Yankee in October and November 1988. The inspectors found that the company failed to keep track of a ring of keys to certain secure areas of the plant. The inspectors also found that Maine Yankee security personnel failed to take expeditious compensatory action once they realized the keys were missing and failed to report the incident to NRC in a timely manner, both required by NRC regulations. The staff proposed a $25 000 fine for this violation.

Other inspection findings included inadequate lighting in parts of the "protected," or controlled access area, of the plant; inadequate monitoring of personnel and packages entering this area; lax control of vehicles within this area; and failure to keep an area adjacent to the protected area, known as the "isolation zone," free of obstructions that might conceal a potential intruder. The staff proposed a $50 000 fine for these violations.

In a letter notifying Maine Yankee of the proposed fine, W. T. Russell, Regional Administrator of the NRC's Region 1, urged company management to focus more attention on the plant's security program. Mr. Russell said, "The NRC recognizes that you developed initiatives. . .for improving the security program. . .Nevertheless, these violations represent serious weaknesses in your program that emphasize the importance of. . .more effective management oversight and attention to the program to assure that security personnel, as well as other individuals authorized access to the plant, understand and adhere to security requirements."

The civil penalty for the key problem was mitigated from $50 000 to $25 000 because of the corrective actions and Maine Yankee's identification of the violation. The civil penalty for the other problems was left at $50 000, notwithstanding the licensee's corrective actions, because the violations were identified by the NRC and the plant security staff should have reasonably discovered the violations sooner, the NRC staff decided.

Overcooling of Reactor Coolant During Trip at Sequoyah 1

The NRC staff proposed a fine of $50 000 against the Tennessee Valley Authority (TVA) as a result of excessively cooled reactor coolant water during three automatic reactor shutdowns (trips) in 1988 (Ref. 26).

Specifically, the Final Safety Analysis Report (FSAR) for the Sequoyah facility requires that, in the event of a reactor trip, the feedwater control system maintain the average temperature of the reactor coolant system at a certain analyzed temperature level to assure that an adequate post-trip margin of safety is maintained.

Contrary to this requirement, during reactor trips on May 19 and 23 and June 6, 1988, the average temperature of the reactor coolant system dropped below the analyzed value. Had the fuel in the reactor core been approaching the end of its useful life, such a condition could have increased the probability and consequences of an accident.

Further, the NRC's regulations require that such deficiencies be promptly identified and corrected. However, the actions taken by the TVA in this regard were not adequately implemented, which resulted in another excessive cooldown following a reactor trip on Nov. 18, 1988.

In addition, the NRC's regulations required the TVA to develop adequate procedures to provide sufficient guidance and acceptance criteria to evaluate plant performance. However, the TVA procedure governing reactor trips did not compare actual post-trip conditions with those set forth in the FSAR. Consequently the post-trip reviews following the May and June reactor trips were inadequate to identify and correct the reactor coolant system overcooling problem.

Together, these three violations constitute a Severity Level III problem, and a fine of $50 000 was proposed.

Radiation Protection Problems at Arkansas Nuclear One

The NRC staff proposed a $25 000 fine against the Arkansas Power and Light Company (AP&L) for alleged violations of NRC radiation protection requirements at Arkansas Nuclear One (ANO) (Ref. 27).

The NRC based this action on inspections conducted Nov. 3–4 and 28–29, 1988, at ANO, after the company informed NRC of apparent overexposures to two workers, one on November 2 and the other on November 23. Although both overexposures violated NRC requirements, only the November 2 incident led to the proposed fine.

Both of these exposures apparently were caused by "hot particles," highly radioactive specks sometimes found in nuclear power plants that can adhere to workers' clothing or skin. The first incident involved a maintenance employee working near a steam generator who received a whole-body radiation dose of 3.2 rem. This brought his quarterly dose to 4.5 rem, which is above the NRC calendar quarter limit of 3 rem.

The second incident affected a painter's helper working near main steam isolation valves. He received a 61-rem exposure to a localized area of his skin; this exceeded the NRC limit of 7.5 rem to the skin in a calendar quarter. This limit, however, is primarily based on the assumption that a large skin area is subject to exposure, rather than a localized area, as was the case at ANO.

Neither exposure is expected to result in discernible, adverse health effects. The NRC is assessing a civil penalty only for the first overexposure, and a lower-level violation for the second, because of the greater radiological significance of a whole-body dose in excess of limits. The NRC also reduced in half the base $50 000 civil penalty for a violation of this sort because of AP&L's good past performance in radiation protection, its detection of these incidents, and its corrective actions.

Physical Security Violations at Sumner

A fine of $62 500 was levied against South Carolina Electric and Gas Company because of alleged violations of physical security requirements at the Sumner plant.[28]

Details of security arrangements at nuclear power plants are exempt from public disclosure. However, NRC officials said the company was notified that the proposed fine was due to what the NRC termed as breakdowns in the access control program.

The NRC said that the base civil penalty for the violations cited against the company was $100 000 but that the amount was partially mitigated as a result of the company's identification of the violation and corrective action taken once the violation was identified.

Five Safe-Operation Violations at Fitzpatrick

The New York Power Authority (NYPA) was cited for five instances of violations of NRC requirements for safe operation of the Fitzpatrick nuclear power plant, which resulted in the imposition of a $75 000 fine.

Three of the violations involved NYPA (1) not declaring an ECCS inoperable, (2) not taking timely corrective action when service water flow through the ECCS room coolers was found obstructed as the result of accumulation of silt in the pipes, and (3) having an inadequate test procedure to find this plugging problem. These room coolers provide necessary cooling to the ECCS equipment during an accident in which the ECCS is required to operate. The NYPA failed to shut down the plant within 24 hr, as would have been required if they had declared the ECCS inoperable during the last operating cycle.

The two remaining violations involved operating the plant while the service water system was at a higher water temperature than specified without completing a written safety evaluation for such operation and without promptly notifying the NRC Operations Center. The service water system removes excess heat from sources other than the reactor, such as pumps, motors, and room coolers. The higher temperature condition was caused by a higher-than-normal temperature of Lake Ontario, the source of water for the service water system.

In a letter to NYPA, NRC Region 1 Administrator W. T. Russell said the alleged violations represented a significant failure to ensure timely and systematic evaluation of plant conditions and demonstrated the need for better coordination and communications within the NYPA organization "to ensure that safety issues are promptly identified and corrected."

Russell said that the plant had operated for about 2 weeks with the high service water temperature limit having been exceeded and that NYPA officials had not notified the NRC Operations Center of this condition within 1 hr, as required.

Although the base civil penalty amount for these violations is $50 000, Mr. Russell indicated that the penalty was increased by 50% because of "multiple examples of a failure to adequately disposition potential safety issues."

Loss of Secondary Containment Integrity at Brunswick 1

The CP&L was hit with a $150 000 fine as a result of two simultaneous events that individually resulted in the loss of secondary containment integrity over a 3-day period in December 1988 (Ref. 29).

The NRC said that, while irradiated fuel assemblies were being moved inside the secondary containment building, an NRC resident inspector discovered that the standby gas treatment system that would be used to process radioactive gas in the event of a mishap was inoperable. A second violation occurred during the same time frame when reactor operators discovered that all four secondary containment isolation ventilation system dampers had been inadvertently isolated on December 9 when an operator failed to do an adequate operations clearance review, which resulted in isolation of the air supply that operated the dampers.

The Notice of Violations noted that regulations require that, any time irradiated fuel is moved, both an operable standby gas treatment system and reactor building ventilation secondary containment isolation dampers are required to be operable so that, in the event a fuel assembly is dropped, any radioactive gases released are processed and released through an elevated vent to minimize any ground level release of radioactivity.

The base civil penalty for each violation is $50 000, the NRC Notice said, but the first violation was increased by 50% because of the Company's failure to identify it and by another 50% for poor past performance in the areas of operator error and attention to detail. The NRC said that a postulated fuel handling accident with the conditions present at the time of these events would result in a minor release of radioactivity within allowable limits but that it was concerned about the broad breakdown of checks and balances that allowed this to happen.

Missing Vortex Suppressors in Containment Sumps at South Texas 1

The NRC staff proposed to fine Houston Lighting and Power Company (HL&P) for an alleged installation error at both Units 1 and 2 of the plant.[30]

The NRC fine stems from HL&P's discovery on Nov. 29, 1988, that required vortex suppressors were missing from the emergency sumps in the plant's two containment buildings. The company then declared the emergency systems that relied on these suppressors inoperable, shut down Unit 1, and installed the devices in both Units 1 and 2. (Unit 2 was not yet licensed to operate). The NRC inspectors confirmed the circumstances and the company's corrective actions.

Vortex suppressors are stainless steel gratings intended for use if an accident causes spilled reactor coolant to collect in the containment building sumps where the devices are installed. Should this water need to be suctioned for recirculation through the emergency cooling system, these gratings would prevent a whirlpool effect that would draw air into piping and pumps and thus reduce the system's effectiveness and possibly damage it.

In a letter informing HL&P of the proposed fine, NRC regional administrator R. D. Martin said the NRC staff concluded that the lack of these vortex suppressors would have caused a degradation, rather than a loss, of the safety systems involved if they had been activated. Mr. Martin also acknowledged that HL&P found and reported the situation to NRC and then took "prompt and extensive corrective action."

He said the company should have known the vortex suppressors were missing. He noted that HL&P included them as a design feature in the South Texas safety analysis report and took credit for them when it certified to NRC that Unit 1 was ready to operate. He further pointed out that the violation had existed for about 8 months before it was discovered in November 1988.

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Recent Developments

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General Administrative Activities

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"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 January, February, and March 1989. Subjects discussed, among others, are Advisory Committee on Reactor Safeguards (ACRS) comments on a variety of issues, the Department of Energy's (DOE's) energy-use predictions for the next decade, and a Nuclear Regulatory Commission (NRC) review of post-TMI actions to comply with the recommendations of the President's Commission on the Accident at Three Mile Island.

ACRS COMMENTS ON SEVERAL ISSUES

During the first quarter of 1989, the ACRS communicated a number of opinions, relating to a variety of issues, in letters to the NRC. Each of these will be briefly discussed and cited in part.

Mark I Containment Improvement Program

The reactor containments, called Mark I, designed by General Electric for some of their boiling-water-reactor (BWR) power reactors have long been the subject of special attention because of concerns that they may be especially vulnerable to failure when called upon to mitigate severe core-melt accidents. Conditional failure probabilities as high as 90% have been estimated for some accident sequences. Even such a high probability may be acceptable, in comparison to other plants, however, if the occurrence probabilities for the sequences for which such high containment failure probabilities are calculated to be sufficiently low and if the core-melt probability for these reactors is lower than for average light-water reactors (LWRs).

The NRC staff has proposed a series of improvements to the Mark I containments intended to achieve the latter end, i.e., lower core-melt occurrence probability without raising the former, conditional containment failure probability subsequent to a core-melt accident.[a] The ACRS considered this matter and arrived at conclusions quite at variance with those of the NRC staff. In its letter to the Commission on this subject, the ACRS said, in part:[1]

> The package of improvements for Mark I plants being proposed by the staff is primarily directed toward lowering the probability of core melt without changing noticeably the conditional probability of containment failure. The staff has documented estimates of a factor of five to ten in the reduction of core melt probability due to internal accident initiators for plants that incorporate the proposed recommendations. Estimates of improvement in containment performance have not been calculated, although there are statements that the probability of failure will be reduced. We were told in an oral presentation that the improvements might reduce the conditional probability of failure to less than 50 percent. It was emphasized that this was only an estimate.
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> We have previously expressed our opinion that the Commission's safety goal is an appropriate standard

[a]See also the Section "NRC Staff Comments on Mark-I Containment Safety" in *Operating U.S. Power Reactors* in this issue of Nuclear Safety.

for establishing how safe plants should be. We also have suggested in our letter dated April 15, 1986, that an implementation plan for the safety goal should provide a framework for assuring that plants have adequate defense in depth as well as assuring that they meet quantitative risk standards. As a class, Mark I plants, as indicated by several PRAs for particular plants, appear to conform to the quantitative risk standards. These plants may not have an appropriate balance between prevention and mitigation. (For this discussion, we define prevention as those activities intended to keep the core from melting, and mitigation as those activities intended to keep fission products released from a melted core away from the public.)

On the basis of a limited analysis of the potential costs and benefits of the proposed improvements, the staff concludes that the improvements are generally cost beneficial and are thereby justified for all 24 Mark I plants. We do not agree. A number of assumptions used in the analysis seem not 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 conclude that no risk-based reason has been identified which justifies singling out Mark I plants from the general population of LWRs. There is a program to look at all plants to identify any possible "risk outliers." This is the Individual Plant Examination (IPE) program. We believe that Mark I plants should be analyzed as a part of this program and that vulnerabilities in individual plants can thereby be identified, analyzed, and corrected where necessary.

We recommend that the proposed improvement plan for Mark I containments be dropped so that licensee and NRC resources can concentrate on the more effective IPE approach.

Proposed Final NRC Rule on Standardization and Licensing Reform

The NRC is considering issuing a ruling on plant standardization and licensing reform which would encourage the former and go far toward simplifying the latter. The proposed rule (10 CFR 52) would provide for early site permits, standard design certifications, and combined construction and operating licenses for nuclear power plants. After considering the issue the ACRS reported to the Commission;[2] its letter states, in part:

Since we have not yet seen the final version of the Draft Final Rule, the public comments, or the Statement of Considerations, our comments below may be subject to revision or amplification after we have seen the final version of these documents.

We recommend that the various types of designs be named and defined more clearly than in the proposed rule. We suggest the following:

- Improved LWR Designs—for LWR plant designs that contain improvements beyond those designs of LWR plants licensed for construction prior to the effective date of this rule.
- Advanced LWR Designs—for LWR plant designs that differ significantly from improved LWR designs or use simplified inherent passive, or other innovative means to accomplish safety functions to an extent significantly greater than in improved LWR designs.
- Advanced Non-LWR Designs—for advanced plant designs using other than light water as moderator or coolant.

The information required for design certification is identified in Section 52.47(a)(2). This section includes a requirement for the submittal of information sufficiently detailed to permit the preparation of procurement specifications and construction and installation specifications. The staff's review of this material can be performed most efficiently and with greater understanding if this large body of information is available in final form, i.e., the procurement specifications and the construction and installation specifications. We recommend that the rule be expanded to require submittal of these documents.

The references in Part 52 to the responsibility of ACRS for review should be made consistent with the provisions of the Atomic Energy Act of 1954, as amended.

We will continue to follow and review the development of this rule along with the Statement of Considerations and advise you accordingly.

One month later the ACRS returned to the same subject since in the interim a more complete draft of the final rule became available to the committee. After reviewing this draft the ACRS submitted the following comments, cited in part:[3]

Section 52.47 b(2)(i) of the draft final rule establishes the requirements for certification of a standard design which differs significantly from an "evolutionary" light water reactor design, or which utilizes simplified, inherent, passive, or other innovative means to accomplish its safety function. We have several concerns with the provisions of this section as written. We interpret this section to provide for the following:

(1) Certification of a design may be granted without testing if the scope of the design is complete and the analysis of the performance and interdependence of the safety features is found acceptable. We recommend against providing for certification of a design solely on the basis of analysis.

The staff indicates that our concerns can be handled by proper modification of the Statement of Considerations.

(2) Certification may be granted for a design whose scope is less than complete if the testing of a prototype demonstrates that the noncertified portion of the plant cannot significantly affect safe operation of the plant. Our problem with this provision is that unless the design of the noncertified portion of the plant is well defined and considered, the potential adverse effects on safe operation of the plant from the noncertified portion may not be identified by testing of the prototype. We recommend against providing certifications for less than complete scope for these designs.

Our letter of January 19, 1989 on the incomplete final rule package included a recommendation for requiring the submittal of procurement specifications and construction and installation specifications as an appropriate indication of the expected scope and level of information required for effective review of an "essentially complete" design. Requirements for design and procurement type specifications did appear in the Standardization Policy Statement of September 15, 1987, but were not included in the draft final rule. We believe they should be.

It is noteworthy that the requirements which we recommend, appear in the Electric Power Research Institute report, "Advanced Light Water Reactor Utility Requirements Document" (June 1986) and in the Atomic Industrial Forum (AIF) report, "Standardization of Nuclear Power Plants in the U.S." (December 16, 1986). The AIF document also states that, "the degree of design detail necessary for providing an "essentially complete" design will generally be that detail which is suitable for obtaining specific equipment or construction bids.

Code Scaling, Applicability, and Uncertainty Methodology for Uncertainty Analysis of ECCS Evaluations

The "Code Scaling, Applicability, and Uncertainty" (CSAU) approach is a methodology developed by the NRC Office of Nuclear Regulatory Research (RES) for determining the overall uncertainty in the calculation of the maximum fuel cladding temperature in the course of a loss-of-coolant accident (LOCA). By late 1988, RES had completed its development effort on the methodology and conducted a demonstration of the method as applied to the realistic (i.e., best-estimate rather than conservative) calculation of the peak cladding temperature (PCT) in a large-break LOCA (Ref. 4). The ACRS has considered the applicability of the CSAU approach and commented as follows:

The objective of developing the CSAU methodology has been to provide a technical basis for quantifying uncertainty in estimates of the PCT expected in a large-break LOCA. Estimates are generated using the realistic analytical models and computer codes permitted under the revised emergency core cooling systems (ECCS) rule. The CSAU methodology is intended to provide not only a practical method for estimating uncertainty, but also one that is well documented and can be audited. This is especially important because calculation of PCT is complex and estimates of its uncertainty require the combination of quantitative analysis and expert opinion.

The CSAU methodology should serve as an appropriate guide for the NRR staff to use in reviewing future submittals from licensees under the revised ECCS rule. It should also serve as a model for methodologies that might be developed and used by licensees and their contractors.

RES has suggested that the general approach used on the CSAU program could be applied to the resolution of issues associated with the NRC's severe accident research effort. We agree.

However, ACRS member H. L. Lewis added the following additional comment:

I have no problem with this letter, except that I believe the Committee has been too charitable toward the claim that this methodology sheds much light on the uncertainty question. Although extensive *sensitivity* analyses were performed as part of this program, sensitivity is not uncertainty. Unless there exists prior knowledge of the uncertainty in the input parameters, sensitivity analyses say nothing about uncertainty. To be sure, comparison of results with experiments would tell something about uncertainty, except that the codes are "matured" by this process. The residual uncertainty for other circumstances is then unknown.

While I applaud this effort, the ultimate question of interest is: "If the calculation predicts a temperature of X°F, what is the chance that it would really be (X+100)°F, *if there were an accident.*" That is uncertainty, and CSAU contributes little to it.

Therefore, the Committee letter may be overenthusiastic in its endorsement of the CSAU methodology for the quantification of uncertainty.

Safety Evaluation Report on the Sodium Advanced Reactor Design

The ACRS has considered the draft Safety Evaluation Report (SER) for the DOE/Rockwell International Sodium Advanced Fast Reactor (SAFR) design. This review is the first SER the

ACRS has had the opportunity to comment upon in an extended period of time and thus marks a resumption, after a fairly extended hiatus, of progress toward approving new reactor designs. The ACRS comments, in part, are as follows:[5]

> The SAFR conceptual design is a product of a DOE program to develop designs for possible future power reactor systems that would have enhanced safety characteristics. Other design projects in the program are the Modular High Temperature Gas Cooled Reactor (MHTGR) and the Power Reactor Inherently Safe Module (PRISM). The NRC staff has reviewed these designs in accordance with the Commission Policy on Advanced Nuclear Power Plants. These preapplication reviews are intended to provide NRC guidance on licensing issues at a relatively early stage of design development. The ACRS has previously commented to you in June 1987 on NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," in July 1988 on key licensing issues associated with the entire program, in October 1988 on the SER for the MHTGR, and in November 1988 on the SER for PRISM.
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> We understand that issuance of the SER will not constitute approval of the SAFR design. Further engineering development and documentation would be required to support a future application for design certification.
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> The SAFR design incorporates small modular reactors cooled by liquid sodium. The standard SAFR plant would consist of one or more "power paks." Each "power pak" would comprise four reactor modules that would produce a total of 3600 MWt (1400 MWe). Each reactor, along with its intermediate heat exchangers and pumps is immersed in a pool of sodium. A steel vessel containing this pool is surrounded by a secondary steel container and each module is installed within a concrete structure above grade. Secondary sodium coolant will flow from each reactor module to a pair of steam generators, located above grade along with the remainder of the balance of plant (BOP) equipment.
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> The SAFR modular design provides several desirable features for enhancing safety of a nuclear power plant:
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> - a passive system for emergency removal of decay power
> - inherent mechanisms for negative feedback of reactivity
> - two independent scram systems, one capable of self-actuation
> - large thermal inertia in the pool of sodium coolant
> - metal fuel, offering greater opportunity for on-site fuel reprocessing
> - small component sizes, providing opportunities for factory fabrication
> - opportunity for prototype testing of a single module
> - separation of safety-related functions from BOP systems
>
> SAFR, while similar to PRISM, has some important differences. Each SAFR reactor module is larger and would generate 900 MWt compared with 425 MWt for PRISM. SAFR primary sodium would run hotter than in PRISM with a nominal core exit temperature of 950°F compared with 875°F for PRISM. SAFR steam conditions are 850°F and 2700 psig, compared with 545°F and 990 psig for PRISM. SAFR has two reactivity control and scram systems while PRISM has one. SAFR's main coolant pumps are conventional centrifugal while PRISM's are electromagnetic.
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> The DOE has decided to discontinue its development of the SAFR design and concentrate liquid metal reactor (LMR) efforts in the PRISM design organization, but has requested that the NRC staff complete its review of both SAFR and PRISM. The NRC staff has expressed no opinion that there appears to be a net advantage in the PRISM design over that of SAFR, or vice versa.
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> On the basis of its review, the NRC staff has concluded that the SAFR design has the potential for a level of safety at least equivalent to current light water reactor (LWR) plants. We have no reason to disagree and believe that SAFR, like PRISM, could be licensed if continuing development work is pursued successfully.
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> A number of safety issues remain to be completely addressed. A continuing program of research and development will be necessary to support further design. Plans for extensive prototype testing should be included. In the following paragraphs, we comment on a number of specific safety issues which we believe should be considered by the staff in its final SER, and by DOE if it continues design and development of this concept.
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> Positive Sodium Void Coefficient
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> SAFR, like PRISM, will experience a large increase in reactivity in the event of significant boiling or other voiding of the sodium coolant. The designers' analyses cannot show that such voiding is impossible, but they have concluded that it is very improbable. Whether it is improbable enough and whether the consequences of such voiding can be tolerated is the major safety issue that must be resolved before these reactor designs could be licensed. The simultaneous and sudden loss of both main circulation pumps, without scram, in a reactor module might cause significant sodium boiling and a reactivity increase. If the positive voiding coefficient is to be accepted, such events must be shown to be of extremely low probability. We believe that additional design and safety analysis work is needed in this area.

Other Reactivity Coefficients

The satisfactory performance of the system in certain low probability transients is very dependent on the changes in core reactivity with variations in power, temperature, and flow that can make subtle changes in the core geometry. For these transients there are small margins between the calculated response and unacceptable responses. A considerable design and development effort will be necessary to assure that response of the core will be acceptable over a wide range of potential challenges.

Scram Systems

The SAFR design includes two sets of control rods either of which can independently shut down the reactor in response to a scram signal and maintain it subcritical. One set would be released automatically by the loss of holding power in a special clutch containing a magnet. Abnormally high sodium temperature, greater than 1050°F, would cause the Curie point temperature of the magnet to be exceeded. We note, however, that this feature depends on there being maintained a sufficient flow of sodium coolant over the magnet. This flow must be assured if the automatic shutdown is to be assured.

Neither of the control rod systems is fully safety grade. Apparently, the systems do have some of the most important features of safety grade systems, e.g., tolerance of single failures. While we agree that experience with LWRs indicates the designation of a system as safety grade is not a guarantee of high reliability, we suggest that designation of a system as fundamentally important as a scram system as non-safety grade is flouting not only convention but good sense.

Use of PRA

The NRC staff seems to have been disappointed in the extent to which PRA has been useful in reviewing the design of SAFR, as well as the earlier review of the PRISM and MHTGR. Apparently the designs at this stage are developed in so little detail that risk analysts have little to work with and the benefits of the analysis are limited. Decision makers should regard with caution quantitative claims of high safety performance for reactor systems still at the conceptual design stage.

Containment

Although a secondary vessel is provided to contain leakage of sodium coolant, the SAFR design does not include a conventional containment capable of resisting high temperatures and pressures. It is contended that the potential for accidents, for which such a containment might provide mitigation, is so low that a conventional containment is not needed. Both deterministic and probabilistic arguments are made in support of this contention. Although these arguments have technical merit, we are not yet convinced. Our position is as stated in our report to you of July 20, 1988 on the key licensing issues associated with DOE-sponsored reactor designs and our report to you of October 13, 1988 on the preapplication safety evaluation report for the Modular High Temperature Gas Cooled Reactor.

However, there is a problem in specifying containment design criteria. One reason for providing a strong physical containment is to protect the public against unforeseen accidents. But, precisely because they are not foreseen, the design requirements for a containment are not obvious. Therefore, engineering and policy judgments must be made about the need for, and nature of, containment that might be used with SAFR. We believe that further study is appropriate before final judgments are made.

Individual Rod Worth

There are two shutdown systems utilized in SAFR. Neither is currently safety grade. The automatic plant trip system can drive in all six of the primary control rods, which have a net reactivity worth of about ten dollars. It can also interrupt power to the electromagnetic latch and drop three secondary control rods, with a net reactivity worth of about seven dollars. The minimum number of primary control rods needed for reactor shutdown is two out of the six to insert about three dollars. The secondary system needs but one rod (about 2.2 dollars) to enter the core. With this very large reactivity worth for each rod, there is a potential for serious consequences from a rod ejection accident. We believe that this requires further study.

Need for Local Flow and Temperature Monitoring

The SAFR safety analysis indicates that blockage of flow through one fuel assembly may damage that assembly, but will not damage adjacent assemblies. Early work with oxide fuel has demonstrated that propagation is unlikely, but experiments and analysis with metal fuel have not been as extensive. Especially because the design does not provide for monitoring flow and effluent temperature from individual assemblies, we believe that this requires further study.

Role of the Operator

We believe that insufficient attention has been given to the role of the operator. Claims that a SAFR plant would have such inherently stable and safe characteristics that the operator will have essentially no safety function are unproven. Operation of four reactors, possibly in several different operational states at any given time, may be a significant challenge for the small operations crew envisioned. Opportunities for cognitive error, which might defeat favorable safety characteristics of the reactor, might be more abundant than is now recognized. Further study is needed.

Other Operational Considerations

In addition, certain features that have been found to be desirable in LWR plants are not provided in the

SAFR design. Although remote shutdown capability is provided, it appears to lack some of the attributes of such systems in current LWR plants. Also, the design does not include Class 1E AC electric power systems, but relies entirely on Class 1E DC power from batteries. We recommend that further consideration be given to the potentially large power needs of essential auxiliary functions such as space cooling.

Protection Against Sabotage

With regard to the need for designing protection against sabotage, the following statement from our report of July 20, 1988 should be given early consideration as the design of this plant progresses:

> It is often stated that significant protection against sabotage can be inexpensively incorporated into a plant if it is done early in the design process. Unfortunately, this has not been done consistently because the NRC has developed no guidance or requirements specific for plant design features, and there seems to have been no systematic attempt by the industry to fill the resulting vacuum. We believe the NRC can and should develop some guidance for designers of advanced reactors. It is probably unwise and counterproductive to specify highly detailed requirements, as those for present physical security systems, but an attempt should be made to develop some general guidance.

Sodium Fires

Further study of the potential for and suppression of sodium fires and consideration of their possible consequences is needed. Such studies should include the possibility of fires resulting from earthquake effects.

Resolution of the Air Systems Reliability Generic Issue

A resolution to Generic Issue 43 "Air Systems Reliability"[6] has been proposed by the NRC staff.[7] The ACRS had the following comments on the proposed resolution:[8]

The instrument air (IA) system commonly is classified as a non-safety system even though it may be the sole source of IA to many safety-related components. The justification for a non-safety classification is that the safety-related components it serves are designed to fail to a safe state if the air pressure is lost. The design and testing of IA systems assume that the loss of air pressure will be instantaneous. Operating experience has shown that gradual loss of air pressure is possible and that, under these conditions, certain components supplied with air by the IA system may behave differently than expected. This constitutes an unreviewed safety issue.

The NRC staff has told us that the gradual loss of air pressure is not addressed in the resolution of Generic Issue 43. More specifically, the staff stated that requirement (3) in Generic Letter 88-14 does not require verification of proper operation of air-operated safety-related components under this condition. Although some studies made by staff contractors suggest that the potential for multiple common-cause failures as a result of improper IA system performance is not a significant contributor to risk for many plants, these results have not convinced us that this finding is correct or can be extended to all plants, which would seem to be a logical requirement if a generic issue is to be resolved.

In view of the above discussion, we do not consider the resolution of Generic Issue 43 as adequate. We support what has been proposed or done by the staff and the industry as described in the resolution package for Generic Issue 43, but further work is needed to show that the gradual loss of air pressure issue is not a safety problem for any plant.

Implementation of the NRC Safety Goal Policy

The ACRS has been studying NRC staff plans for implementing the NRC Policy Statement on Safety Goals[9] and has twice before issued comments on the staff plans.[10,11] In early 1989 they again addressed this subject in a letter to the Commission which reads, in part, as follows:[12]

Although we agree with the general direction of the staff's recommendations, we have substantive differences about a number of issues. We urge the Commission to implement the policy after considering our recommendations.

Background

The [NRC staff's] draft paper proposes guidelines for the NRC staff to use in implementing the Safety Goal Policy. These guidelines include the structure of an implementation plan, definitions, and quantitative objectives. The paper calls for these guidelines to be incorporated into the policy statement itself through an amendment. In addition, the paper proposes that potential averted on-site costs be used as an offset to licensee costs in cost–benefit analyses. And finally, the paper asks the Commission itself to consider whether the policy should be amended to clarify the relationship of the safety goal and the statutory standard of adequate protection.

Before commenting specifically on the staff paper, an observation about the use of probabilistic risk assessment (PRA) and its relation to the safety goal is appropriate. Although it is frequently said that "the bottom line is the weakest part of PRA," the fact remains that the safety goal cannot be implemented without the bottom line. Without this bottom line and a safety goal to which it can be compared, either explicitly or implicitly, PRA becomes a never-ending search for outliers. Although it is satisfying to some engineers and analysts to identify "dominant" contributors to risk, especially those that can be eliminated readily, there is nothing necessarily less safe about a plant that has most of its risk embodied in one or two outlier sequences than a plant that has its risk distributed more or less uniformly over 20 sequences.

Structure of the Implementation Plan

The draft paper describes a structure similar to that suggested in our letter of May 13, 1987, but with some differences. We continue to prefer the structure we recommended, a hierarchical arrangement of five levels using the multiple goals in the policy statement of August 6, 1986.

The staff's current proposal is consistent with our recommendations for Levels One and Two. Level One is the pair of qualitative goals and Level Two is the two quantitative health objectives.

Our recommendation for Level Three would be the general performance guideline that large accidental releases should occur no more frequently than 1E-6 per reactor-year. The staff's Level Three proposal is similar, but differs in the definition of "large release."

The staff proposal defines a large release as "a release that has a potential for causing an offsite early fatality." We are still not satisfied with this definition for two reasons. First, it can or could be considered as little more than the quantitative health objective in Level Two, but at a level ten times more conservative. Second, this considerable additional conservatism is not accompanied by a significant simplification. The use of the word "potential" in order to encompass the release at Chernobyl will require the use of Level 3 PRA results with a suitable prescription or selection of potential meteorology and population distribution or location. Although this would be possible for specific plants, it would require arbitrary assumptions if the safety goals are to be used to test the sufficiency of the Commission's regulations or to provide a basis for establishing design criteria for containments for future plants.

We continue to believe that a definition in terms of the release itself is preferable. It might be defined in terms of curies, leak or release rate, or fraction of the core or containment inventory. In any case, it should be independent of the site characteristics and should provide some criteria against which the design or performance of containments can be tested. We urge you to request the staff to continue seeking a means to define a large release that is not significantly more conservative than the Level Two health objectives and that focuses the mitigative function on containment design characteristics independent of site or population characteristics.

Our recommendations for Level Four consisted of three specific performance objectives: (1) core melt probability, an expression of the effectiveness of a plant's *prevention* systems, (2) conditional probability of containment failure, an expression of the effectiveness of a plant's *mitigation* systems, and (3) an expression of how well a plant is operated. (We use here the term "prevention" to describe those activities and systems intended to keep the reactor core from melting, and "mitigation" to describe those activities and systems intended to keep away from the public fission products that would be released from a melted core.) Level Four proposed by the staff is significantly different from what we recommended. It would consist of only one of the three objectives we recommended, a limit on core damage frequency. This loses the balance between prevention and mitigation, one form of defense-in-depth, that is inherent in our inclusion of a containment performance objective. We believe this balance should be retained.

The staff proposal for Level Four also omits the ACRS recommendation for a quantification or objective statement of how well a plant is operated. We called this a "plant performance objective." We have not been able to develop a workable definition for this, nor has the staff. In light of this, we rely upon the alternative recommendation made in our letter of April 12, 1988: "If this cannot be done, a prominent caveat, e.g., a warning that PRA results do not tell the full story, should be made a part of the policy or of the implementation plan." We recommend that such a statement be made an explicit part of the plan.

In our letter of May 13, 1987, we recommended a quantitative objective of 1E-4 per reactor-year for "core melt" as a part of the Level Four performance objectives. In our letter of April 12, 1988, we more carefully defined the event that should be associated with this quantitative objective as the "loss of adequate core cooling (core overheating beyond design-basis limits)." The staff proposal seems to agree with our recommendation. We caution, however, that comparisons of this objective with some of those proposed by others under the description of core melt probability can be misleading.

We disagree with the staff's proposal to use 1E-5 per reactor-year as the target for mean core damage frequency for future plants. This difference from the objective for existing plants introduces an arbitrary level of conservatism which conflicts with the criterion we suggested for linking the hierarchical levels of safety goal objectives; that is, that each subordinate level of the hierarchy should be consistent with the level above and should not be so conservative as to create a *de facto* new policy. Not only would the staff proposal introduce a major inconsistency with the Level Two and Three objectives, but it would result in loss of balance between prevention and mitigation because arguments could then be made that the higher levels of the safety goal hierarchy could be met readily without the need for accident mitigation systems such as containment buildings. The Commission's safety goal should be the same whether considering the adequacy of regulations for existing plants or for future designs, and whether for LWRs or other types of reactor plants.

Definition of "Adequate Protection"

The term "adequate protection" has importance in the legal areas of safety regulation. Although it is needed and used with apparent precision in legal instruments, its technical definition is not precise. In general, it is accepted as equivalent to the term "with no undue risk to public health and safety" often used in other contexts. Another term, "in full compliance with the regulations" is used as a surrogate, on occasion, for either of these.

We believe that the safety goal should play an important, but indirect, role in defining adequate protection. Ideally, compliance with the Commission's regulations is a suitable surrogate for defining adequate protection of the public. However, we believe that the adequacy of the regulations should be judged from the viewpoint of whether nuclear power plants, as a class, licensed under those regulations, meet the safety goals. It is our understanding, following discussions with the staff, that the staff proposes the safety goal to be a sort of aspirational objective which would be sought but not necessarily reached.

With the safety goal approach now proposed by the staff, a class of plants that meets existing regulations (therefore meeting a standard of adequate protection) would be obliged to make improvements up to the safety goal, if cost–benefit arguments so dictated. The implementation plan proposed earlier by the staff would have used the safety goal as the minimum standard (i.e., adequate protection) and cost–benefit arguments could have been used to justify further improvements, without other limits. We believe that neither of these approaches is a proper use of the Safety Goal Policy.

We believe that the proper use of the safety goals is embodied in two principles which we have previously recommended:

(1) The safety goal is a definition of how safe is safe enough.

(2) At the present time, the safety goal should be applied to judging the adequacy of regulations and regulatory practices, and not to make specific decisions about individual plants.

The Commission has taken a bold and progressive step in proclaiming the Safety Goal Policy. It is an attempt to place the regulation of safety in nuclear power plants in an appropriate context relative to other risks in society. It is imperfect, but it is as useful a step as has been taken by any industry or regulatory agency. Using concepts of cost–benefit analysis or, even worse, ALARA (as low as reasonably achievable), dilutes the achievement and effectiveness of the Safety Goal Policy. We believe that the safety goal is a good present standard for "how safe is safe enough." Further, as we have stated earlier, we believe that the safety goals should be used to judge the adequacy of the regulations from the standpoint of whether those regulations result in classes of nuclear power plants which can be and are operated in such a way as to meet the safety goals, and thus provide adequate protection to the public.

A wide community of safety experts and policy makers has concurred, after extended deliberation, in accepting the Safety Goal Policy as reasonable, based on present knowledge. It may be that future information about reactor risk or societal risk will cause a need to adjust the safety goal one way or another, or to make different implicit allowance for uncertainty. Until that happens, we believe that the safety goal should be accepted as an unambiguous working standard for the regulation of nuclear power, along the lines we have suggested.

Cost–Benefit Analysis

The staff paper proposes that cost–benefit analyses made to evaluate proposed plant safety improvements should use averted on-site costs as an offset to the plant costs entailed in making such improvements. We believe that this is appropriate in making cost–benefit assessments, although it inevitably adds uncertainty to the results. However, as discussed above and as we stated in our letter of April 12, 1988, we believe cost–benefit analysis is not properly a part of safety goal implementation (in contrast to "backfit" implementation).

Incorporation of Guidelines Into the Policy

We concur with the staff proposal to incorporate certain of the implementation guidelines as amendments to the policy statement. We have no preferences or comments about the details of this, beyond the reminder that the safety goal is a policy statement, not a regulation.

Coherence Among Regulatory Policies

The Safety Goal Policy has been in existence for some time and has, in fact, been an influence in recent regulatory activities. We believe a clear implementation plan is necessary to ensure that it is applied comprehensively, consistently and unambiguously. Several major Commission decisions are presently on the horizon regarding, for example, the Severe Accident Policy, the issue of Mark I containment adequacy, certification of advanced reactor designs, and evaluation of plant operations. In each of these, the question "how safe is safe enough" must be answered, either implicitly or explicitly. The safety goal can and should bring greater objectivity, consistency and clarity to deliberations and decisions about these issues.

Generic Issue 99: Improved Reliability of RHR Capability in PWRs

On a prior occasion[13] the ACRS commented on a proposed resolution of Generic Issue 99 relating to improving the residual heat removal (RHR) capability in PWRs. After reviewing the NRC staff response to these comments, the ACRS submitted additional thoughts that read, in part:[14]

> This generic issue addresses concerns about the possible failure of core cooling during shutdown operations in PWR plants. Analyses have shown there is a significant risk of core damage from overheating due to the loss of RHR circulation from a number of possible causes. The leading cause of this risk, as indicated by both analysis and experience, is loss of core cooling as a result of errors made during so-called "mid-loop" operation. The staff issued a generic letter that identifies a number of actions licensees are advised to take to reduce the likelihood of such incidents. The generic letter also recommends that licensees develop certain appropriate procedures and equipment that will permit rapid closing of any containment openings in such emergencies so that, if core damage does occur, release of fission products from the containment will be minimized.
>
> In our September 14 report, we expressed agreement with most of the recommendations of the staff's generic letter, but questioned whether the plan for emergency closure of containment openings had been sufficiently analyzed, given the many varieties of containment openings that exist in actual plants. In our recent discussions with staff members, we learned that they intend to conduct inspections in all plants for compliance with recommendations of the generic letter. These inspections will be carried out by the resident inspector staff at each plant and will be supplemented by more in-depth inspection conducted by specialists from the headquarters staff for selected plants.
>
> We are particularly interested in the conclusions about the effectiveness of the proposed containment closure procedures that will be drawn as a result of these inspections. We want the staff to brief us within a few months, and we will be especially interested in information about the nature of the containment closures involved. For example, some closures are designed so that pressure within the containment will tend to compress closure seals. Other closures are designed so that pressure will tend to decompress and perhaps open gaps in containment seals. It is apparent that the proposed procedures for rapid installation of closures will be more effective in achieving containment for those with the former type of seal than with the latter.
>
> A more general policy issue, which should be considered by the Commission rather than the staff, is apparent in the GI 99 resolution. The staff has presented estimates to us which show that the risk caused by loss of RHR cooling under shutdown conditions is a significant fraction of the total risk from reactor operation. Despite this, resolution of GI 99 is being carried out by informal means, through recommendations in a generic letter, rather than by more formal means, e.g., through rulemaking. We believe the staff's approach probably will be effective, and we have no quarrel with it. However, we question why more formal methods, ultimately more burdensome to licensees and staff, are used in resolution of other issues less important from the risk standpoint. There does not appear to be any well-defined policy direction from the Commission concerning which regulatory approach should be taken for a given circumstance.

Generic Issues 70 (PORV and Block Valve Reliability) and 94 (Additional Low-Temperature Over-Pressure Protection for LWRs)

The ACRS concurred in the proposed resolution of these two generic issues, conditional upon the addition of certain clarifications to the Plant Technical Specification Action Statements· in Enclosures C-1, D-1, and E to the draft generic letter in the proposed resolution package.[15] The concurrence letter states the proposed clarifications as follows:

> 1) When one or more block valves associated with power operated relief valves (PORVs) are closed because of excessive relief valve seat leakage, it should be required that electrical power be maintained to the block valves to ensure quick reopening capability from the control room. This requirement was discussed in the Staff's Regulatory Analysis (Section 5.2) but was not stated explicitly in the Modified Technical Specifications. We believe it should be.
>
> 2) In the Surveillance Requirements section, the staff should state that the reactor coolant system should be in hot shutdown rather than cold shutdown when performing an operability test on the block valves or PORVs. In the Regulatory Analysis the staff states that stroke testing of these valves should be performed only at cold shutdown. During our discussions with staff members, they agreed that hot shutdown is the correct requirement.
>
> 3) The Surveillance Requirements section should also include the solenoid air control valves and check valves on associated air accumulators. The inservice testing requirement stated in the Staff Regulatory Analysis does include valves in PORV control systems. We believe that this statement should be modified to clearly specify the solenoid valves and the accumulator check valves in PORV control systems.

ACRS member W. Kerr added the following additional comments:

Although intuitively I believe that improving the performance of power operated relief valves would decrease risk of reactor power plant operation, I do not believe the Staff's Regulatory Analysis demonstrated that this would occur. Nor do I believe it showed that what is proposed would improve the performance of relief valves.

ACRS members H. L. Lewis and P. G. Shewmon jointly added the following remark:

We were told by the staff that studies show that these valves pose an insignificant risk, but that, for other reasons, that were not presented, they disagree with the analysis. That makes this an example of regulation for the sake of regulation with little impact on safety. As such, it is a bad example. We think that they have done no harm, but that is an inappropriate standard for the resolution of generic issues.

Proposed Severe Accident Research Program Plan

The ACRS was asked to comment on the NRC staff's proposed Severe Accident Research Program Plan, dated February 1989, prior to the submission of that plan to the Commission. Stating that the time constraints made it impossible to perform a detailed review, the ACRS nevertheless made a preliminary assessment, on the basis of which they communicated the following letter (in part) to the Commission:[16]

The NRC began the Severe Accident Research Program shortly after the TMI-2 accident. The emphasis was said to be on understanding severe accident phenomena, and in developing a capability to calculate the risks of severe accidents. Computer codes were expected to play a key role in these calculations, and development of these codes and experiments related to their validation have represented a significant part of the severe accident research. Our previous reviews of the program have frequently led us to question the relevance of this research to regulatory needs. As a result, we have written a number of reports to the Commission recommending that there be a closer correlation between the severe accident research proposed and the policy being formulated to ensure protection of the public from the risk of severe accidents. We saw much of the severe accident research as not properly focused to provide the information needed.

In contrast, the February 1989 program plan proposes a review of the information available from previous research to identify areas in which further information is needed for regulatory decisions. Existing and proposed research programs will be reviewed and, if necessary, redirected to make it more likely that the needed information will be developed. It is also proposed that a method of evaluation, such as Code Scaling, Applicability, and Uncertainty recently developed by the staff for analysis of thermal–hydraulic codes, be used to evaluate a number of the severe accident codes. Further, in light of the fact that there appears to be duplication among some of the severe accident codes under development, it is proposed to examine which of these codes are needed for regulatory applications, and on the basis of the results, to decide which codes deserve further development. It is also proposed that documentation be required for both existing codes and those under development.

On the basis of our preliminary review, we believe that this program plan represents a substantial change and is a very positive step. We endorse the staff's requirement that all contractors show that their proposed and continuing work address analyses or phenomena important in the predictions of risk, and have clearly defined objectives. We recommend that the Commission encourage the staff to continue in the direction indicated. Because this represents a significant departure from previous practice, some parts of the program are likely to encounter opposition. It is important that this be monitored carefully to ensure that it does not deter the positive aspects of the proposed program.

We expect to continue our review. However, our initial examination leads to the following specific observations.

The near-term program dedicates a major fraction of the total resources to studies of various phenomena associated with direct containment heating (DCH). We believe that as an alternative, a greater priority should be given to studies that might very well demonstrate that risk from DCH is negligibly low, or could be made low by readily achievable plant modifications or procedural changes, thus making much of the proposed DCH related research unnecessary.

The draft plan we have does not indicate how results of previous work or expected results from existing research programs of U.S. industry or foreign organizations are to be factored into the NRC program. We expect to explore this further.

Additional Applications of LBB Technology

The ACRS made comments on considerations by the NRC regarding a proposed policy statement on additional applications of leak-before-break (LBB) technology.[17] The ACRS comments read, in part:

The central concept of . . . (LBB) involves acceptance of the argument that, in a given piping system, small leaks through cracks in pipe walls can be detected before the cracks have grown to a size where they can cause a sudden gross failure of the pipe. Further, the argument says that when the leak is detected, the damaged pipe will be taken out of service before the crack has had a chance to grow to a size that is on the threshold of unstable propagation. In 1987, the NRC revised General Design Criterion 4 (GDC 4) to permit the use of the LBB concept for certain purposes and under certain circumstances in both existing and new nuclear power plants. This revision made it possible for licensees to exclude the dynamic effects of hypothetical sudden pipe ruptures from consideration in the design of certain pipe support structures, if the piping systems in question met certain conditions.

In granting its approval for the GDC 4 revision, the Commission recognized that there is nothing inherent in the LBB concept that limits the application to the use specified and stated that, "There are possibly other areas which could benefit from expanding the leak-before-break concept and simplification of requirements such as environmental qualification and ECCS." In response, the staff solicited public comments on this subject through a notice in the *Federal Register* dated April 6, 1988. A range of opinions was cited in 23 comment letters. After considering these comments, the staff recommended that no rulemaking be undertaken to apply the LBB concept to either ECCS or environmental qualification. They pointed out that any safety benefits associated with the application of the LBB concept to ECCS can be more readily obtained under the recently revised ECCS rule. In addition, the broad scope revision to GDC 4 permitted the use of exemptions for applying LBB to environmental qualification.

In our discussions with the NRC staff, it became apparent that they believe the potential safety enhancements that might result from extending the LBB concept would not be great enough to justify the large expenditure of resources needed to develop bases for rulemaking. They seemed to feel that the industry's failure to use the exemption option in the existing rule indicated a lack of industry interest. The staff indicated that requests for exemptions, suitably documented and supported, might eventually provide the basis for a rule extending the LBB approach to environmental qualification.

In presentations to the ACRS, some representatives of the industry expressed their belief that there was a real potential for substantial safety and/or economic benefits in applying the LBB concept to both ECCS and environmental qualification. However, they were reluctant to expend their own resources on activities that they felt would not lead to changes in the rules.

We agree with the staff's conclusions to the extent that rulemaking at this time would be premature. However, we believe an avenue for consideration of further extension of the LBB concept should exist. As a result of our most recent discussions of this issue with the staff and with industry representatives, we believe that the staff is open to a serious consideration of industry proposals to extend the concept to situations for which technical justification can be provided. We recommend that the policy statement contain language which makes it clear that this is the case.

International Agreement to Study TMI-2 Pressure Vessel Lower Head

On Jan. 31, 1989, the NRC entered into an agreement with 10 other countries to cooperate in investigating the condition of the lower head of the TMI-2 reactor vessel.[18]

The investigation of the reactor vessel damaged in an accident in 1979 will take 3 yr and cost an estimated $7 million. The ten countries, under the auspices of the Organization for Economic Cooperation and Development's Nuclear Energy Agency (NEA), will share up to 50% of the cost of the project sponsored by the NRC. The countries are Belgium, Finland, France, Federal Republic of Germany, Italy, Japan, Spain, Sweden, Switzerland, and the United Kingdom.

Advanced cutting tools developed under the agreement will be used to remove a minimum of 8 samples and possibly as many as 20 from the bottom head of the reactor vessel. The samples will be examined in U.S. laboratories and in some of the other participating countries.

Data obtained from testing the samples will be used to determine the thermal and chemical effects on the reactor vessel and the amount of structural integrity remaining. During the accident, molten material was quenched in the vessel's lower region after coming into direct contact with the inner vessel surface; this demonstrated the existence of additional margins of safety in a severe accident.

The results of this research are important to determine the extent of the safety margin for use in the evaluation of severe accidents and to develop improved accident management methods for existing light-water reactors.

Data from testing the samples will also be integrated with TMI-2 research studies of some of the core debris by DOE to gain a better understanding of core-melt sequences. More than 200 000 lb of core debris have been removed in the cleanup, which was expected to be completed later this year.

NRC Staff Proposes Accident Management (AM) Regulations

On the basis of Probabilistic Risk Assessments (PRAs) and severe accident analyses, the NRC staff has concluded that the risk of core damage in severe accidents can be significantly reduced through effective AM (Ref. 19). The staff then identified several generic AM strategies that, in their view, can enhance a licensee's ability to cope with the accidents that are risk-dominant in the PRAs. In a meeting on Jan. 23, 1989, the staff requested the Commissioners to approve the implementation of an AM program.

According to the proposed policy paper discussed at the meeting, AM encompasses actions taken by a plant's staff to: (1) prevent core damage, (2) terminate core damage should it begin and retain the core within the reactor vessel, (3) maintain containment integrity for as long as possible, and (4) minimize offsite releases. In response to a question from Chairman Zech, however, the staff made it clear that in this context AM refers only to severe accidents greater than a plant's design basis accident.

As part of the proposed program, each NRC licensee would be expected to implement an AM plan to provide a framework for evaluating information on severe accidents—including information gathered from individual plant examinations—to prepare and implement severe accident operating procedures and to train plant personnel in these procedures.

According to the proposed policy, licensees will be expected to make available to the plant's staff guidance for diagnosing the progress of severe accidents and planning an appropriate response. In addition, they will be required to make any changes in instrumentation necessary to carry out their AM procedures. Finally, it is anticipated that each individual plan will include a review and/or modification of the plant's current decision-making authority.

In its presentation, the staff acknowledged that AM plans exist at all U.S. reactors to varying degrees. They stated, however, that a more disciplined approach is necessary. The proposed AM plan would primarily seek to make the most effective use of existing plant resources—both personnel and hardware. In the personnel category, examples would include defining the chain of command during a severe accident and ensuring that each operator knows his or her role. Although AM might involve using plant hardware in unanticipated ways, such as using the fire control apparatus to help cool the core, the NRC staff proposal asserts that major hardware changes are not a central aim of the plan.

The Commissioners seemed generally receptive to the concept of severe accident management. There was substantial dispute during the briefing, however, over many of the specific points in the staff's proposal. Commissioner Roberts indicated that he felt the staff's proposals amounted to new requirements and should be formally treated as such.

Commissioner Carr was the most outspoken critic of the proposal. He cited among his concerns the possibility that preparing numerous emergency responses to unlikely severe accidents would actually increase the chances of unintentionally creating an accident. Carr also took issue with the staff's decision to recommend that water always be added to the core during the course of a severe accident. Carr concluded his comments by suggesting that the NRC staff first sit down and try to comply with the new requirements before sending them to the utilities.

In his concluding remarks, Commissioner Rogers seconded Carr's concern regarding an increased potential for accidents. Rogers was also apprehensive about moving ahead with the AM proposal before the individual plant evaluations are complete.

"Safety Culture" Concept Promulgated in New NRC Policy Statement

On Jan. 24, 1989, a Final Policy Statement[20] on the conduct of nuclear power-plant operations was issued in the *Federal Register.*[21] The policy statement, effective immediately upon publication, states that the working environment of a plant has a direct impact on safety, and plant managers have an obligation to promote a "safety culture" inside the plant. In the statement, NRC cited a number of occasions on which it received reports of operator inattentiveness or unprofessional behavior.

The NRC used the International Nuclear Safety Advisory Group's definition of "safety culture" to mean the "personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants." To promote this "safety culture," NRC expects appropriate behavior in each nuclear control room.

The policy statement cites specifics of such behavior as follows:

1. Conduct within the control room should always be professional. The operator at the controls, as well as his supervisor, must never relinquish his position unless to a qualified operator, and not without a thorough turnover briefing.
2. Operations within the control room should be performed with formality. Appropriate consideration should be given to the need for acknowledgment and verification of verbal commands.
3. The control room must be secure from intrusion. In addition, only qualified personnel must be allowed to manipulate the controls.
4. The operator must be continuously alert to plant conditions and activities that could affect plant operations; all alarms should be responded to promptly.
5. Only such activities necessary to the operation of the plant should be permitted in the control room.
6. Activities outside the control room, such as on-line maintenance, should be fully coordinated with the control room.
7. Detailed written records of plant operations must be kept at all times.
8. The control room environment should minimize unnecessary distractions.
9. Foreign objects not needed for plant operation should be restricted from the control area to avoid unintentional activation of the controls.

The Commission intends this policy statement to make clear NRC expectations and to provide guidance for meeting these expectations. The NRC said it believes that utility management should routinely monitor control-room operations to ensure that they meet these guidelines.

DOE Issues Energy Forecast for Next Decade

The DOE's Energy Information Administration (EIA) issued its "Annual Energy Outlook—The Long-Term Projections"[22] in late January 1989; the report extends its projections to the year 2000 (Ref. 23). For the past year, the report states, energy use in the United States increased by 3.4%. Continued growth, albeit at a somewhat lower rate, is forecast to continue.

Between 1988 and 2000, EIA projects electricity sales to expand by about 2.6% per year. The report predicts that in 2000 total domestic energy production will be 69.3×10^{15} Btu, up slightly from EIA's last estimate of 68×10^{15} Btu. The EIA says that nearly all the change can be attributed to a higher estimate for natural gas production.

The EIA concludes that, because most announced new generating facilities will come on line by the mid-1990s, there will be a sharp need for additional capacity through the remainder of the century. Most of this new capacity will come from natural gas-fired combined-cycle plants, the report says.

In the first nine months of 1988, "Energy Outlook" reports, the utilization rate of nuclear plants was 64.5% as compared to 57.7% in 1987. In addition, four of the eight units that became operational in 1987 had utilization rates greater than 85% in July 1988. The report notes that this is considerably higher than the typical capacity factor for reactors in their first fuel cycle. According to EIA, these higher rates seem to be a result of increased electricity demand related to extreme weather, as well as the utilities' emphasis on improved plant performance through increased operator training, and less frequent refueling outages.

On the whole, however, the EIA report is not optimistic about the future of the nuclear energy industry. The report assumes that, even though legislation to streamline the nuclear licensing process was introduced in the 100th Congress, no new plants will be ordered before 2000. Nuclear power generation is forecast to grow only by an annual rate of roughly 0.8% for the 1988–2000 period. (By comparison, natural gas is expected to grow by 6.5% annually.) The 0.8% growth figure translates into a total of 6.2×10^{15} nuclear-produced British thermal units in 2000.

Nevertheless, the report does add that these projections could be affected by the reactivation of deferred nuclear construction projects. A total of 36 units, each more than 1% complete, have been deferred or canceled. Thirty-one of these are beyond resurrection. Five others—Grand Gulf-2, Perry-2, Seabrook-2, and Washington Public Power Supply System Projects 1 and 3—have been deferred and could conceivably be reactivated if market conditions made it worth while. The report says that Grand Gulf-2, Perry-2, and WPPSS-1 and 3 could be operable in four and a half to seven years following a reactivation decision.

Senate Testimony on Anti-Energy Terrorism

On Feb. 8, 1989, the U.S. Senate Government Affairs Committee heard testimony on the vulnerability of the U.S. energy infrastructure to terrorism.[24]

First to address the committee was R. K. Mullen, an independent consultant on energy sabotage issues. He began with a description of the antinuclear terrorism that plagued West Germany in 1986. He said that there were assassinations of nuclear industry officials, attacks on power lines, and a wide-ranging sabotage campaign against the infrastructure supporting nuclear power. This included attacks on construction firms, banks, and transportation facilities.

Since 1986, however, Mullen said the level of sabotage has declined by about 80%. The decrease is due in part to frustration with the lack of concrete results plus added security at potential targets. Mullen believes that the antinuclear terrorism in West Germany is merely a subset of a larger antitechnology campaign.

For the United States, Mullen painted a much brighter picture. Although serious acts of sabotage have been reported, Mullen pointed out that the most frequent incidents involve shooting at power lines and bombing gas stations. In addition, many of the incidents occur in conjunction with strikes or other labor disputes.

One incident involving nuclear facilities occurred in February 1980 at the Browns Ferry Unit 2 reactor. Mullen told the committee that on three separate occasions employees intentionally tripped the reactor and thus forced the Tennessee Valley Authority to buy $3 million in replacement power.

Another episode involving a nuclear facility occurred in May 1986 when unknown individuals grounded 500-kV power lines leading from the Palo Verde reactor complex in Arizona. The reactors were not operating at the time. If they had been, the grounding would have caused them to shut down automatically.

In general, Mullen told the committee, most acts of energy sabotage in the United States are directed less at facilities than at those who own or operate them. He saw no indication that this trend was likely to change, even given the presence of terrorist support groups in the United States.

A second viewpoint on the impact of terrorism came from C. H. White of the National Electrical Manufacturers Association. White focused on the manufacturing industry's ability to help utilities recover from widespread damage to large power transformers.

White's testimony portrayed a significant decline in the heavy electrical equipment manufacturing industry. He painted a picture of an industry which lost 40% of its production capacity in the last two and a half years and which faces unfair competition from subsidized foreign manufacturers.

In response to questions from Senator Glenn, White contended that there is only one U.S. plant capable of making high-voltage transformers and that each transformer takes a year or more to produce. When asked if terrorist attacks could make it impossible to replace these generators, White said "yes."

The question of what actions have been taken by the electric utilities to combat terrorism was addressed by M. R. Gent, president of the North American Electric Reliability Council (NERC). Gent said that, although he did not dispute the importance of the issue, he had serious reservations about testifying in an open hearing. Consequently he only provided a sanitized outline of what has been done to meet the threat of terrorism.

Gent said that since 1983 NERC has worked quietly with the government to design a program to strengthen national electric security preparedness. He applauded the relationship as "very impressive" and said that some security proposals had already been implemented whereas others were under consideration.

Under questioning from Senator Glenn, Gent insisted that the industry has contingency plans for events, such as hurricanes, that are much more severe than terrorist threats. In a follow-up question, Sen. J. Lieberman (D-Connecticut) asked what assumptions the NERC security group had made about terrorism. Gent replied that it had assumed terrorism will occur even though many of the group members were skeptical of such a conclusion. Gent went on to stress that no multi-source attacks, those most likely to cause widespread damage to energy facilities, have been recorded in developed countries.

The committee heard about utility security procedures from D. B. Hinman, manager of corporate security for the Alabama Power Company. Comparing the electrical system to a spider web, Hinman said that the system's reliability was based largely on its redundancy features—one broken strand would not cause the entire web to fall.

Although Hinman said he was aware of the potential terrorist threat, he was not particularly complimentary when responding to questions regarding DOE's "threat advisories." He accused the advisories of being so general as to be of "questionable value."

Threat advisories were discussed again during the day's final presentation by Deputy Assistant Secretary for Energy Emergencies E. V. Badolato. Badolato told Senator Glenn that there are differences between security advisories and security alerts. Alerts are put out by the Federal Bureau of Investigation and are plant or region specific. The DOE, on the other hand, issues nonspecific advisories that it gleans from State Department notices sent to U.S. firms operating overseas. Badolato agreed that there were problems with several advisories issued in 1987 and 1988 but he said that the warnings do fill a vacuum. However, he declined to recommend either greater government activity or greater funding for the energy emergency section of DOE.

Pro-Nuclear "National Energy Policy Act of 1989" Introduced in Senate

Senator T. E. Wirth (D-Colorado) introduced S. 324, the National Energy Policy Act of 1989, in February 1989 (Ref. 25). The stated goal of the bill is to reduce CO_2 emissions to 20% of 1988 levels by the year 2000, a truly ambitious undertaking. The bill requires the Secretary of Energy to establish a passively safe reactor research program and, in addition, prepare a comprehensive report on the status of producing electricity from thermonuclear fusion.

The bill specifies that the reactor research and development program should attempt to develop technologies that permit modular design, exhibit passive safety, are adaptable to standardized construction, are cost-effective compared to other energy sources, minimize nuclear waste, prevent diversions of material to nuclear weapons, and minimize the cost of plant decommissioning. $500 million is authorized for research between 1991 and 1993.

The legislative proposal also envisions a report by the Secretary of Energy outlining what steps have been taken to ensure a demonstration of magnetic and inertial fusion ignition by the year 2010. If the report concludes that commercial fusion is feasible, the secretary is expected to develop a prototype fusion reactor along with cost estimates sufficiently accurate to allow construction bidding.

S. 324 has 30 cosponsors and was referred to the Committee on Energy and Natural Resources.

NRC Proposes Amendments to 1954 Atomic Energy Act

The NRC has sent copies of a draft bill to amend the 1954 Atomic Energy Act (AEA) to the Speaker of the House and the President of the Senate.[26] In an accompanying letter, NRC Chairman Zech listed the purposes of the proposed legislation, which consists of a series of amendments to the AEA. The purposes are to:

1. Help ensure that regulatory violations and defects in components are reported to NRC.
2. Clarify NRC's right to protect safeguards information.
3. Eliminate congressional reporting requirements for the Advisory Committee on Nuclear Safeguards.
4. Establish an Advisory Committee on Nuclear Waste.
5. Permit guards at NRC licensed facilities to carry firearms.
6. Help deter the introduction of weapons into NRC facilities.
7. Help deter sabotage.
8. Modify the procedures for initiating prosecutions under the AEA.
9. Authorize NRC to obtain administrative search warrants.

Firms constructing or supplying the components of nuclear plants would be directly responsible for notifying NRC of any defects. This would include DOE when its facilities were NRC-licensed. The NRC would be allowed to conduct investigations or other enforcement activities to assure compliance with the regulations.

Concerning safeguards information, NRC says it has noticed "ambiguities" in current law that might permit generic information to endanger site-specific safeguards measures implemented by licensees. The amendments would clarify NRC's power to withhold such generic information. According to NRC, similar legislation was passed by the House and Senate last year but not enacted into law.

Under NRC's proposed amendments, the ACRS would no longer have to report annually to Congress on nuclear safety research. NRC claims that members of the ACRS are only part-time government employees, and preparation of the report is not an efficient use of their time. The ACRS would, however, continue to report on specific projects.

Security personnel at NRC facilities with special nuclear materials would be allowed to use deadly force to prevent the theft of such material. The NRC notes that DOE guards already have this authority. For the moment, the amendment would apply only to facilities making fuel for navy nuclear propulsion. In the future, however, it might be extended to enrichment or reprocessing plants.

In another security-related proposal, NRC would be allowed to produce regulations prohibiting persons from carrying weapons or explosives into any Commission-licensed facility. In the draft bill NRC said it intends to limit the prohibition to those facilities which must be protected against theft or sabotage. During the 100th Congress, this provision was passed by the House but not the Senate.

In addition, acts of sabotage at NRC-licensed facilities under construction would be considered federal crimes. Current law applies only to completed plants. Again, the House passed similar legislation but the Senate did not.

Presently, the attorney general must personally approve the initiation of prosecutions under the AEA. The proposed amendments would remove that requirement, which would uncomplicate internal Department of Justice procedures and allow more timely prosecutions, according to NRC.

Finally, access to administrative search warrants would permit NRC to inspect, without prior notice, the premises of a firm that the Commission believes may be in violation of regulations that could affect public health and safety. Existing law gives NRC clear authority to inspect licensees, but it is less clear whether inspections of nonlicensed firms that impact the nuclear industry are permitted. The NRC says that, although this "grey area" hampers its ability to protect public health and safety, it would use its new powers only when absolutely necessary.

The draft legislation would still have to be introduced by a member of Congress. There was no indication when, or if, that would happen.

FEMA Publishes Guidelines for Emergency Response in Case of State Nonparticipation

On Nov. 18, 1988, then-President R. Reagan issued Executive Order 12657 requiring the Federal Emergency Management Agency (FEMA) to prepare emergency plans for the areas surrounding nuclear plants should state and local governments fail to do so. In the February 28 *Federal Register* (54 FR 8512), FEMA published an interim rule describing how it would comply with that order.[27] The rule was developed by a task force of FEMA and NRC staff.

Should a locality decline or fail to provide an emergency response plan, FEMA is required to: assist the licensee in the development of such a plan, participate in the testing of the plan to ensure its effectiveness, and prepare for, and if necessary undertake, an operational role in an emergency. However, FEMA may not substitute its resources for those of local governments except to the extent necessary to make up for their nonparticipation. Under the "realism doctrine," FEMA assumes local governments will make their best effort to protect the public during an emergency, including requesting federal assistance.

Once FEMA determines that a state or locality is not participating in emergency nuclear planning, the agency will attempt to supply the licensee with the resources necessary to establish a workable proposal. Under an earlier agreement, FEMA will then evaluate the licensee's emergency preparedness and convey that information to NRC. NRC will, in turn, use the information in its licensing and regulatory procedures.

In the event of an actual radiological emergency, FEMA's rule says that the agency will coordinate with state and local governments to take whatever action is needed to deal with the crisis. FEMA will be the lead agency in directing responses but will transfer its functions to local entities when they begin to exercise their authority.

The FEMA is authorized to receive reimbursement for its activities from licensees and nonparticipating local governments. However, the procedures for reimbursement have not yet been established and were to be published in a future *Federal Register.*

Earlier in March 1989, Congressman E. J. Markey (D-Massachusetts) urged President Bush to rescind the Reagan executive order under which FEMA acted in issuing these guidelines.[28] It is Markey's view, contained in a letter to the President, that, on the basis of a Congressional Research Service analysis of the Executive order, FEMA has no authority to assume "command and control" functions at the state and local level, as provided in the order. President Bush was reportedly studying the Markey letter and was to respond when his study was completed.

Senate Bill Introduced to Create Independent Nuclear Safety Board

On Mar. 15, 1989, Sen. J. R. Biden (D-Delaware), with the cosponsorships of Senators A. Gore (D-Tennessee), J. Kerry (D-Massachusetts), H. Metzenbaum (D-Ohio), and D. Pryor (D-Arkansas), introduced a bill, S.589, to create a Nuclear Safety Board (NSB) independent of the NRC (Ref. 29). The bill asserts that there is a "great need" for intensive investigation of mishaps at NRC-licensed facilities as well as continual need for critical review of NRC's regulatory practices.

The bill provides for the establishment of a three-member board that would be empowered to conduct investigations at NRC-licensed facilities or to ask NRC to conduct the inquiries in any instance that it feels is a potential threat to public safety. Under the bill, an investigation is justified if the NSB determines that NRC has acted or failed to act in a way that presents a danger to the public health. On the basis of these investigations, it would then suggest regulatory changes to the Commission and Congress. The NRC would then have 4 months to either adopt the recommendations or explain why it chose to reject them. In addition to investigations of specific events, the Board would be free to conduct special safety studies at NRC-licensed facilities.

To carry out its investigations, the Board would have access to operational data from all NRC-regulated facilities. It would also be free to analyze NRC data, including personnel files. To further assist its functions, the Board would establish reporting requirements for anyone involved in an NRC-licensed activity, whether it be the operation of a nuclear plant or the transportation of nuclear material. S. 589 also would incorporate the functions of NRC's Office for the Analysis and Evaluation of Operational Data into the NSB.

The results of NSB investigations or studies would be available to the public.

For fiscal years 1990 to 1995, the bill would authorize $6 million annually for the NSB. After 1995 the Board would be disbanded. A spokesman for Biden said this was consistent with the senator's philosophy of putting "sunset" dates on his legislation. He also indicated that after 6 yr of oversight by the NSB, NRC might become sufficiently safety conscious so that the Board would no longer be needed.

The legislation was referred to the Committee on Environment and Public Works. Although similar legislation passed the Senate during the 100th Congress as part of an NRC reauthorization bill, it was not clear when or how the committee would act in 1989.

$20 000 Fine Proposed for Whistle-Blower Discrimination

On Mar. 14, 1989, the NRC staff proposed a $20 000 fine against General Electric Company (GE) for discriminating against an employee at GE's Wilmington, N.C., fuel-fabrication facility who took operational safety concerns to the NRC (Ref. 30).

Beginning in 1982 and continuing into 1984, Ms. V. English reported safety concerns about operations in the facility's Chemet Lab. to the attention of the NRC and GE management. As a result, she was removed from her job in the laboratory, and her employment was terminated on July 30, 1984.

On August 24 of that year, Ms. English filed a complaint with the Department of Labor alleging discrimination under provisions of the Energy Reorganization Act of 1974. Her complaint was upheld by the Administrator of the Wage and Hour Division of the Department of Labor and, on appeal, by an Administrative Law Judge. Thereafter the Secretary of Labor and the Court of Appeals dismissed the Labor Department proceeding because of the failure of Ms. English to file a timely complaint.

After reviewing the decision of the Administrative Law Judge, the staff determined that a violation of NRC requirements occurred since, under Part 70 of the Commission's regulations, GE is prohibited from discriminating against its Wilmington employees for bringing safety concerns to the attention of NRC. Accordingly, this incident has been categorized as a Severity Level II violation (Level I being the most serious and Level V the least serious category of violations of NRC requirements).

General Electric Company was given 30 days to pay the fine or protest its imposition in whole or in part. The company also was required to advise the NRC in writing of the steps it has taken or was taking to assure that similar violations would not occur in the future.

In a separate action, the NRC staff granted in part and denied in part Ms. English's request for the NRC staff to take certain actions against General Electric. Her request that a civil penalty be imposed was granted but not in the amount she proposed; however, her request that the NRC staff impose a condition on GE's Wilmington license requiring the company to reimburse her for past and future economic losses was denied.

NRC Report Summarizes Post-TMI-2 Actions

In late March 1989 the NRC issued a report summarizing the status of the actions taken in response to the recommendations of the Presidential Commission on the Accident at Three Mile Island.[31] The report comes just 10 yr after the March 28, 1979, accident.

In announcing the report, NRC Chairman Zech said:

> The NRC staff has completed a review of the progress during the last decade on recommendations of the Presidential Commission following the Three Mile Island accident. Clearly significant modifications and improvements have been made in NRC's and the industry's organization and practices. Training, equipment, and maintenance at our commercial nuclear power plants have been upgraded. Emergency planning has been enhanced. Noteworthy progress has been achieved in adding to the margin of safety already in place in commercial nuclear power reactors. There is a heightened safety commitment within the NRC and the nuclear industry.
>
> We have strong evidence from operational data over the last few years that the performance of commercial nuclear plants continues to improve. Our indicators show, that per operating reactor, the number of significant operating events has continued to decrease; the number of unplanned scrams (automatic shutdowns) has declined; the number of safety system actuations has decreased, and the radiation exposure to plant personnel has continued to decrease. Recently capacity factors—a measure of electricity produced—have begun to increase. These indicators, taken together, demonstrate conclusively that the performance of U.S. commercial operating reactors is improving and these reactors are being operated safely. While we are pleased with these trends, we believe there is room for more improvement and that excellence across the board has not been totally achieved. We should be encouraged by these results but not complacent or overconfident. We believe that both the NRC and the industry must remain vigilant to assure that our commercial nuclear power plants continue to operate in such a manner that the benefits of nuclear energy will be supplied safely and efficiently.

The executive summary of the report is cited, in part, here:

> . . .This report, NUREG-1335,. . .follows the sequence of recommendations in the Presidential Commission report and, where appropriate, reflects the commitments and agreements contained in. . .[the NRC's initial response to the report of the President's Commission, NUREG-0632, "NRC Views and Analysis of the President's Commission on the Accident at Three Mile Island"]. The status of ongoing actions not yet complete is reported for reference purposes.
>
> The Presidential Commission found many then-current NRC and industry practices inadequate and in need of improvement. As a result of their report and other TMI-2 studies, the NRC established a number of new programs and initiatives and modified others. The following highlights many of these actions in terms of the broad areas for improvement identified in the Presidential Commission recommendations.
>
> A. NRC Organization and Management
>
> The NRC has reorganized and adopted several new measures to strengthen its management accountability and to place higher priority attention on the safety of plant operations. The NRC has consolidated the majority of its staff in a single location in Rockville, Maryland, to enhance more efficient decision making and to bring the Commissioners and the staff elements responsible for operational safety into close proximity. NRC has restructured the licensing and inspection functions to reflect the shift in the nuclear industry from construction to operation; and established a separate Office of Enforcement to implement a strengthened enforcement policy.

The NRC has also initiated a number of new programs intended to ensure an improved oversight of licensee performance. These include the systematic assessment of licensee performance program, the diagnostic evaluation program, and the performance indicator program. The NRC inspection program has been expanded, particularly through the use of team inspections and by locating resident inspectors at each site. In addition, the Research program has been redirected to place greater emphasis on severe accidents and risk studies. These efforts have provided NRC management more detailed knowledge of plant operating characteristics and daily operational events.

B. Utility and Suppliers

The industry has established the Institute of Nuclear Power Operations, the Nuclear Safety Analysis Center, and the Nuclear Management and Resources Council to aid licensees to improve plant performance and safety. Both the NRC and the industry have placed a high priority on understanding the lessons of operational experience, particularly with regard to root causes, and communicating these lessons to all plants, both domestic and foreign. NRC has verified that responsibilities for plant operations and related plant procedures are clearly defined for both normal and emergency conditions.

C. Training of Operating Personnel

All licensees have extensively revised their training programs for licensed and non-licensed operators. National accreditation of these training programs is now accomplished under close NRC monitoring. A systems approach to training has been established to improve the effectiveness of training programs for plant personnel. NRC operator examinations now focus on a knowledge of plant operations, and the passing grade was increased. More stringent initial operator candidate screening and medical evaluations were instituted. Licensees were required to have a simulator facility and to provide comprehensive training in the diagnosis of and recovery from possible plant malfunctions and potential accident conditions.

D. Technical Assessment

Since the TMI-2 accident, control room instrumentation and layouts have been reviewed against needed capability to mitigate accidents, and plants have been modified as necessary. Inadequacies in plant design and hardware have been corrected. A Safety Parameter Display System has been provided for critical plant parameters to enhance operators' understanding of the plant's safety status. In-depth and comprehensive studies have been, and continue to be, conducted on severe accident and core melt phenomena, plant equipment performance and reliability, and human performance. Detailed risk assessment research activities and studies have characterized potential safety issues. NRC requirements for oral and written reports of operating events have been substantially revised, and a comprehensive operational experience assessment and feedback program has been established. Finally, the TMI-2 accident recovery program has been conducted in a deliberate manner with full documentation to aid in accident modeling and design studies for advanced reactors.

E. Worker and Public Health and Safety

In order to promote increased attention to public health and safety, the NRC has worked to achieve coordination of Federal radiation effects research; adequate training of state and local emergency response personnel; and upgrading of licensee, State, and local radiological emergency response capabilities. Federal radiation effects research and related matters are now coordinated through the Committee on Interagency Radiation Research and Policy Coordination. Licensees are required to provide training to emergency response personnel, and both NRC and the Federal Emergency Management Agency provide additional training directly to such personnel. Further, NRC emergency preparedness regulations have been extensively revised since the TMI-2 accident. Facility modifications have been required; periodic emergency response drills are evaluated; and licensee emergency plans, facilities, training, and equipment are routinely inspected. All licensees now maintain radioprotective drugs for onsite emergency workers.

F. Emergency Planning and Response

Revised NRC rules and guidance have been issued to provide for improved capability for a wide range of accidents. State and local authorities have upgraded emergency plans, equipment, and training and participate with licensees in biennial response exercises. Public notification and information channels have been established and tested. Responsibilities and cooperative procedures with other agencies have been documented and demonstrated through two Federal Field Exercises involving licensee organizations and Federal, State, and local officials.

G. Public Right to Information

To disseminate prompt and accurate information about emergency conditions, a Joint Public Information Center will be established near the site of any future accident. These centers have the necessary facilities to support the media, and will be staffed by Federal, State, local, and utility representatives who can speak authoritatively about the emergency. Arrangements have been established for announcements over the Emergency Broadcast System to disseminate information. NRC ensures that the public is informed of events which are not emergencies through open meetings and widely disseminated documents, and on any release of radioactivity to the environment in excess of NRC limits. Media training is provided on nuclear safety and related subjects.

In summary, since the TMI-2 accident, significant modifications and improvements have been made in

NRC's and the industry's organization and practices. Training, equipment, and maintenance at nuclear power plants have been upgraded. Emergency planning has been enhanced. Noteworthy progress has been achieved in improving the margin of safety inherent in commercial nuclear power reactors. There is now a heightened safety awareness within the NRC and the nuclear industry, and an improved understanding of the lessons of experience taught by this accident.

REFERENCES

1. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Mark I Containment Performance Improvement Program,* Jan. 19, 1989, cited in NRC Press Release 89-16, Jan. 25, 1989.
2. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Draft Final Rule on Standardization and Licensing Reform, 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,"* Jan. 19, 1989, cited in NRC Press Release 89-16, Jan. 25, 1989.
3. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Final Rule on Standardization and Licensing Reform, 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,"* Feb. 15, 1989, cited in NRC Press Release 89-32, Feb. 24, 1989.
4. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Code Scaling, Applicability, and Uncertainty (CSAU) Methodology for Determination of Uncertainties Associated with the Use of Realistic ECCS Evaluation Models,* Jan. 19, 1989, cited in NRC Press Release 89-16, Jan. 25, 1989.
5. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Design,* Jan. 19, 1989, cited in NRC Press Release 89-16, Jan. 25, 1989.
6. Generic Letter 88-14, *Instrument Air Supply System Problems Affecting Safety-Related Equipment,* Aug. 8, 1988.
7. Memorandum from Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, to Victor Stello, Jr., Executive Director for Operations, *Resolution of Generic Issue 43, Air Systems Reliability,* Sept. 30, 1988.
8. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Resolution of Generic Issue 43, Air Systems Reliability,* Jan. 19, 1989, cited in NRC Press Release 89-16, Jan. 25, 1989.
9. NRC Issues Safety Goal Policy, *Nucl. Saf.,* 28(1): 126 (January–March 1987).
10. ACRS Comments on Safety Goal Policy Implementation Plan, *Nucl. Saf.,* 28(4): 571 (October–December 1987).
11. ACRS Comments on a Variety of Safety-Related Issues—Comments on Safety Goal Implementation Policy, *Nucl. Saf.,* 29(4): 530 (October–December 1988).
12. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Further ACRS Comments on Implementation of the Safety Goal Policy,* Feb. 16, 1989, cited in NRC Press Release 89-32, Feb. 24, 1989.
13. Comments on Proposed Resolution of GI 99 Relating to Reliability of RHR Capability in PWRs, *Nucl. Saf.,* 30(1): 135 (January–March 1989).
14. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Further ACRS Comments on Proposed Resolution of Generic Issue 99, "Improved Reliability of RHR Capability in PWRs,* Feb. 16, 1989, cited in NRC Press Release 89-32, Feb. 24, 1989.
15. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Proposed Resolution of Generic Issues 70, "Power Operated Relief Valve and Block Valve Reliability," and 95, "Additional Low-Temperature Over-Pressure Protection for LWRs,"* Feb. 16, 1989, cited in NRC Press Release 89-32, Feb. 24, 1989.
16. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Proposed Severe Accident Research Program Plan,* Mar. 15, 1989, cited in NRC Press Release 89-49, Mar. 17, 1989.
17. Letter from ACRS Chairman F. J. Remick to NRC Chairman L. W. Zech, Jr., *Additional Applications of Leak-Before-Break Technology,* Mar. 14, 1989, cited in NRC Press Release 89-49, Mar. 17, 1989.
18. NRC Press Release 89-19, Jan. 31, 1989.
19. *At. Energy Clearing House,* 35(4): 5 (Jan. 27, 1989).
20. Policy Statement on the Conduct of Nuclear Power Plant Operations, 10 CFR 50 and 10 CFR 55, Jan. 24, 1989, *Fed. Regist.,* 54(14): 3424.
21. *At. Energy Clearing House,* 35(4): 6 (Jan. 27, 1989).
22. Energy Information Administration, *Annual Energy Outlook,* available from Superintendent of Documents, U.S. GPO, #061-003-00605-1.
23. *At. Energy Clearing House,* 35(5): 4 (Feb. 3, 1989).
24. *At. Energy Clearing House,* 35(6): 3 (Feb. 10, 1989).
25. *At. Energy Clearing House,* 35(6): 6 (Feb. 10, 1989).
26. *At. Energy Clearing House,* 35(9): 9 (Mar. 3, 1989).
27. *At. Energy Clearing House,* 35(9): 12 (Mar. 3, 1989).
28. *At. Energy Clearing House,* 35(9): 8 (Mar. 3, 1989).
29. *At. Energy Clearing House,* 35(12): 3 (Mar. 24, 1989).
30. NRC Press Release 89-43, Mar. 14, 1989.
31. NRC Press Release 89-50, Mar. 21, 1989.

Reports, Standards, and Safety Guides

By D. S. Queener

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 (January, February, and March 1989) 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 88-73, Supplement 1 *Direction-Dependent Leak Characteristics of Containment Purge Valves,* February 27, 1989, 2 pages plus 6 pages of attachments.

NRC IN 88-86, Supplement 1 *Operating with Multiple Grounds in Direct Current Distribution Systems,* March 31, 1989, 3 pages plus 2 pages of attachments.

NRC IN 89-01 *Valve Body Erosion,* January 4, 1989.

NRC IN 89-02 *Criminal Prosecution of Licensee's Former President for Intentional Safety Violations,* January 9, 1989.

NRC IN 89-03 *Potential Electrical Equipment Problems,* January 11, 1989, 3 pages plus one-page attachment.

NRC IN 89-04 *Potential Problems From the Use of Space Heaters,* January 17, 1989, 2 pages plus one-page attachment.

NRC IN 89-05 *Use of Deadly Force by Guards Protecting Nuclear Power Reactors Against Radiological Sabotage,* January 19, 1989, 3 pages plus one-page attachment.

NRC IN 89-06 *Bent Anchor Bolts in Boiling Water Reactor Torus Supports,* January 24, 1989.

NRC IN 89-07 *Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Which Render Emergency Diesel Generators In-operable,* January 25, 1989.

NRC IN 89-08 *Pump Damage Caused by Low-Flow Operation,* January 26, 1989.

NRC IN 89-09 *Credit for Control Rods Without Scram Capability in the Calculation of the Shutdown Margin,* January 26, 1989.

NRC IN 89-10 *Undetected Installation Errors in Main Steam Line Pipe Tunnel Differential Temperature-Sensing Elements at Boiling Water Reactors,* January 27, 1989, 3 pages.

NRC IN 89-11 *Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cable Sizing,* February 2, 1989, 3 pages plus 3 pages of attachments.

NRC IN 89-12 *Dose Calibrator Quality Control,* February 9, 1989.

NRC IN 89-13 *Alternative Waste Management Procedures in Case of Denial of Access to Low-Level Waste Disposal Sites,* February 2, 1989.

NRC IN 89-14 *Inadequate Dedication Process for Commercial Grade Components Which Could Leak to Common Mode Failure of a Safety System,* February 16, 1989, 2 pages plus one-page attachment.

NRC IN 89-15 *Second Reactor Coolant Pump Shaft Failure at Crystal River,* February 16, 1989, 2 pages plus one-page attachment.

NRC IN 89-16 *Excessive Voltage Drop in DC Systems,* February 16, 1989.

NRC IN 89-17 *Contamination and Degradation of Safety-Related Battery Cells,* February 22, 1989, 3 pages plus one-page attachment.

NRC IN 89-18 *Criminal Prosecution of Wrongdoing Committed by Suppliers of Nuclear Products or Services,* February 22, 1989.

NRC IN 89-19 *Health Physics Network,* February 23, 1989, 3 pages plus one-page attachment.

NRC IN 89-20 *Weld Failures in a Pump of Byron-Jackson Design,* February 24, 1989, 2 pages plus one-page attachment.

NRC IN 89-21 *Changes in Performance Characteristics of Molded-Case Circuit Breakers,* February 27, 1989, 2 pages plus one-page attachment.

NRC IN 89-22 *Questionable Certification of Fasteners,* March 3, 1989, 2 pages plus 2 pages of attachments.

NRC IN 89-23 *Environmental Qualification of Litton-Veam CIR Series Electrical Connectors,* March 3, 1989, 2 pages plus one-page attachment.

NRC IN 89-24 *Nuclear Criticality Safety,* March 6, 1989, 3 pages plus one-page attachment.

NRC IN 89-25 *Unauthorized Transfer of Ownership or Control of Licensed Activities,* March 7, 1989, 4 pages plus one-page attachment.

NRC IN 89-26 *Instrument Air Supply to Safety-Related Equipment,* March 7, 1989, 3 pages plus one-page attachment.

NRC IN 89-27 *Limitations on the Use of Waste Forms and High Integrity Containers for the Disposal of Low-Level Radioactive Waste,* March 8, 1989.

NRC IN 89-28 *Weight and Center of Gravity Discrepancies for Copes-Vulcan Air-Operated Valves,* March 14, 1989, 3 pages plus 4 pages of attachments.

NRC IN 89-29 *Potential Failure of ASEA Brown Boveri Circuit Breakers During Seismic Event,* March 15, 1989.

NRC IN 89-30 *High Temperature Environments at Nuclear Power Plants,* March 15, 1989, 3 pages plus one-page attachment.

NRC IN 89-31 *Swelling and Cracking of Hafnium Control Rods,* March 22, 1989, 3 pages plus one-page attachment.

NRC IN 89-32 *Surveillance Testing of Low-Temperature Overpressure Protection Systems,* March 23, 1989, 3 pages plus one-page attachment.

NRC IN 89-33 *Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs,* March 23, 1989.

NRC IN 89-34 *Disposal of Americium Well-Logging Sources,* March 30, 1989.

NRC IN 89-35 *Loss and Theft of Unsecured Licensed Material,* March 30, 1989, 4 pages plus 2 pages of attachments.

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 GPO. However, most of the NUREG series documents can be ordered from the Superintendent of Documents, U.S. Government Printing Office (GPO), 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.

AEOD/E902 *Fires and Explosive Mixtures Resulted From Introduction of Hydrogen into Plant Air Systems,* March 31, 1989, 8 pages.

AEOD/T902 *Technical Review Report, Inadvertent Reactor Trips Due to RCS Flow Instrumentation Maintenance Activities,* L. M. Padovan, January 1989, 10 pages.

AEOD/T903 *Generic Implication of Browns Ferry Fire on November 2, 1987,* March 28, 1989, 3 pages.

AEOD/S901 *Special Study Report Maintenance Problems at Nuclear Power Plants,* U.S. NRC Office for Analysis and Evaluation of Operational Data, February 1989, 30 pages.

NUREG-0020, Vol. 12, No. 12 *Licensed Operating Reactors Status Summary Report, Data as of 11-30-88,* January 1989 (GPO).

NUREG-0020, Vol. 13, No. 1 *Licensed Operating Reactors Status Summary Report, Data as of 12-31-88,* February 1989 (GPO).

NUREG-0020, Vol. 13, No. 2 *Licensed Operating Reactors Status Summary Report, Data as of 1-31-89,* March 1989 (GPO).

NUREG-0020, Vol. 13, No. 3 *Licensed Operating Reactors Status Summary Report, Data as of 2-28-89,* March 1989 (GPO).

NUREG-0090, Vol. 11, No. 2 *Report to Congress on Abnormal Occurrences April–June 1988,* December 1988, 33 pages (GPO).

NUREG-0090, Vol. 11, No. 3 *Report to Congress on Abnormal Occurrences July–September 1988,* January 1989, 30 pages (GPO).

NUREG/CR-2000, Vol. 7, No. 11 *Licensee Event Report (LER) Compilation for Month of November 1988,* December 1988, 92 pages (GPO).

NUREG/CR-2000, Vol. 7, No. 12 *Licensee Event Report (LER) Compilation for Month of December 1988,* January 1989, 81 pages (GPO).

NUREG/CR-2000, Vol. 8, No. 1 *Licensee Event Report (LER) Compilation for Month of January 1989,* February 1989, 123 pages (GPO).

NUREG/CR-2000, Vol. 8, No. 2 *Licensee Event Report (LER) Compilation for Month of February 1989,* March 1989, 159 pages (GPO).

NUREG/CR-3950, Vol. 5 *Fuel Performance Annual Report for 1987,* W. J. Bailey and S. Wu, Pacific Northwest Laboratory, Wash., March 1989 (GPO).

OTHER BOOKS AND REPORTS

During January, February, and March 1989, 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, Vol. 10 *A Compilation of Reports of the Advisory Committee on Reactor Safeguards, 1988 Annual,* April 1989, 170 pages (GPO).

NUREG-1340 *Regulatory Analysis for the Resolution of Generic Issue 99: Loss of RHR Capability in PWRs,* A. H. Spano, February 1989, 65 pages (GPO).

NUREG-1355 *The Status of Recommendations of the President's Commission on the Accident at Three Mile Island. A Ten-Year Review,* March 1989, 93 pages (GPO).

NUREG/CR-4792, Vol. 1 *Probability of Failure in BWR Reactor Coolant Piping, Vol. 1: Summary Report,* G. S. Holman and C. K. Chou, Lawrence Livermore National Laboratory, Calif., March 1989 (GPO).

NUREG/CR-4792, Vol. 2 *Probability of Failure in BWR Reactor Coolant Piping, Vol. 2: Pipe Failure Induced by Crack Growth and Failure of Intermediate Supports,* T. Lo et al., Lawrence Livermore National Laboratory, Calif., March 1989 (GPO).

NUREG/CR-4948 *Technical Findings Related to Generic Issue 23: Reactor Coolant Pump Seal Failure,* C. J. Ruger and W. J. Luckas, Jr., Brookhaven National Laboratory, N.Y., March 1989 (GPO).

NUREG/CR-5042, Supplement 2 *Evaluation of External Hazards to Nuclear Power Plants in the United States. Other External Events,* C. Y. Kimura and P. G. Prassinos, Lawrence Livermore National Laboratory, Calif., February 1989, 50 pages (GPO).

NUREG/CR-5078, Vol. 2 *A Reliability Program for Emergency Diesel Generators at Nuclear Power Plants. Maintenance, Surveillance, and Condition Monitoring,* E. V. Lofgren et al., Sandia National Laboratories, N. Mex., December 1988, 65 pages (GPO).

NUREG/CR-5115 *A Review of Boiling Water Reactor Water Chemistry. Science, Technology, and Performance,* M. J. Fox, Argonne National Laboratory, Ill., February 1989, 61 pages (GPO).

NUREG/CR-5116 *Survey of PWR Water Chemistry,* J. Gorman, Argonne National Laboratory, Ill., February 1989, 128 pages (GPO).

NUREG/CR-5245 *A Review of the Crystal River Unit 3 Probabilistic Risk Assessment. Internal Events, Core Damage Frequency,* N. A. Hanan and D. R. Henley, Argonne National Laboratory, Ill., January 1989 (GPO).

NUREG/CR-5250, Vols. 1-8 *Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains. Methodology, Input Data and Comparisons to Previous Results for Ten Test Sites,* D. L. Bernreuter et al., Lawrence Livermore National Laboratory, Calif., January 1989 (GPO).

NUREG/CR-5270 *Assessment of Seismic Margin Calculation Methods,* R. C. Murray et al., Lawrence Livermore National Laboratory, Calif., March 1989 (GPO).

Other Items

NSAC 132 *Determining Reactivity Values for a PWR Natural Circulation Transient,* EPRI Nuclear Safety

Analysis Center, R. D. Mosteller et al., March 1989 [EPRI Research Reports Center (RRC), Box 50490, Palo Alto, Calif. 94303].

NSAC 133 *Analysis of a PWR Natural Circulation Transient Using RETRAN-02,* P. J. Jensen and J. L. Westacott, March 1989, 45 pages (EPRI RRC).

NSAC 134 *Analysis of a PWR Natural Circulation Transient Using ARROTTA,* M. J. Anderson et al., March 1989, 33 pages (EPRI RRC).

NSAC 135 *Validating ARROTTA for a PWR at Beginning of Core Life,* R. D. Mosteller et al., January 1989 (EPRI RRC).

NSAC 141 *Lead Plant Application of Leak-Before-Break to High Energy Piping,* W. L. Server et al., January 1989, 145 pages (EPRI RRC).

NSAC 142 *The Feasibility of Gas Turbines for Alternate AC Power at Nuclear Power Plants,* T. E. Duffy et al., January 1989, 20 pages (EPRI RRC).

***Chernobyl: Law and Communication,* Philippe J. Sands, April 1988, 340 pages, Grotius Publications Ltd., Cambridge, CB3 9BP, U.K.**

Report No. 43 *Radiation Protection: Radiological Protection Criteria for the Recycling of Materials from the Dismantling of Nuclear Installations,* Commission of the European Communities, November 1988, 55 pages (Commission of the European Communities, Directorate Nuclear Safety, DG XI/A/1, Jean Monnet Bldg. C4/49, L-2920, Luxembourg).

Proceedings of an NEA Workshop on Excavation Response in Geological Repositories for Radioactive Waste, held in Winnipeg, Canada, April 26–28, 1988, OECD Nuclear Energy Agency, 1989, 535 pages (OECD Publications and Information Center, 2001 L St., NW, Suite 700, Washington, DC 20036-4095).

Radiation Detection and Measurement, Second Edition, Glenn F. Knoll, 1989, 755 pages, John Wiley and Sons, Inc., New York ($63.95).

In-Plant Practices for Job-Related Health Hazards Control, Volume One, Production Processes, L. V. Cralley and L. J. Cralley (Eds.), January 1989, 935 pages, John Wiley and Sons, Inc., New York.

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.

Single copies of draft guides may be obtained from NRC Distribution Section, Division of Information Support Services, Washington, DC 20555.

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

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 (GPO), 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, DC, 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 January, February, and March 1989 reporting period, are listed below.

Division 1 Power Reactor Guides

1.158 *Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants,* February 1989.

Division 3 Fuels and Materials Facilities Guides

3.44 (Revision 2) *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type),* January 1989.

3.61 *Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask,* February 1989.

3.62 *Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks,* February 1989.

Division 7 Transportation Guides

7.8 (Revision 1) *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material,* March 1989.

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 January, February, and March 1989. 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.

ANSI B16.41-1983 (R1989, reaffirmation, approved by ANSI/BSR) *Functional Qualification Requirements for Power-Operated Active Valve Assemblies for Nuclear Power Plants.*

ANSI N303-1978 (Withdrawn, approved by ANSI/BSR) *Guide for Control of Gasborne Radioactive Materials at Nuclear Fuel Reprocessing Facilities.*

BSR N300-1975 (Withdrawal of ANSI N300-1975, for comment) *Design Criteria for Decommissioning of Nuclear Fuel Reprocessing Plants,* $6.00.

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 2.13-1979 (R1988, reaffirmation, approved by ANSI/BSR) *Evaluation of Surface-Water Supplies for Nuclear Power Plant Sites.*

ANSI/ANS 3.3-1988 (Revision of ANSI/ANS 3.3-1982, approved by ANSI/BSR) *Security for Nuclear Power Plants.*

ANSI/ANS 8.1-1983 (R1988, reaffirmation, approved by ANSI/BSR) *Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors.*

ANSI/ANS 8.6-1983 (R1988, reaffirmation, approved by ANSI/BSR) *Safety in Conducting Subcritical Neutron Multiplication Measurements in Situ.*

ANSI/ANS 8.10-1983 (R1988, reaffirmation, approved by ANSI/BSR) *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement.*

ANSI/ANS 15.4-1988 (Published) *Selection and Training of Personnel for Research Reactors,* $45.00.

ANSI/ANS 19.1-1983 (R1989, reaffirmation, approved by ANSI/BSR) *Nuclear Data Sets for Reactor Design Calculations.*

ANSI/ANS 19.3-1988 (R1989, reaffirmation, approved by ANSI/BSR) *The Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors.*

ANSI/ANS 19.3-4-1976 (R1989, reaffirmation, approved by ANSI/BSR) *The Determination of Thermal Energy Deposition Rates in Nuclear Reactors.*

ANSI/ANS 19.4-1976 (R1989, reaffirmation, approved by ANSI/BSR) *A Guide for Acquisition and Documentation of Reference Power Reactor Physics Measurements for Nuclear Analysis Verification.*

ANSI/ANS 56.4-1983 (R1988, reaffirmation, approved by ANSI/BSR) *Pressure and Temperature Transient Analysis for Light Water Reactor Containments.*

ANSI/ANS 58.11-1983 (R1989, reaffirmation, approved by ANSI/BSR) *Cooldown Criteria for Light Water Reactors.*

BSR/ANS 8.17-1984 (Reaffirmation of ANSI/ANS 8.17-1984, for comment) *Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors,* $17.00.

BSR/ANS 8.19-1984 (Reaffirmation of ANSI/ANS 8.19-1984, for comment) *Administrative Practices for Nuclear Criticality Safety*, $13.00.

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.

ANSI/ASME NQA-2c-1988 (Published) *Quality Assurance Requirements for Nuclear Power Plants.*

Institute of Electrical and Electronics Engineers

Standards prepared by IEEE can be obtained from IEEE, Attention: M. Lynch, 345 East 47th Street, New York, NY 10017.

ANSI/IEEE 833-1989 (New standard, approved by ANSI/BSR) *Recommended Practices for the Protection of Electric Equipment in Nuclear Power Generating Stations from Water Hazards.*

ANSI/IEEE 845-1988 (New standard, approved by ANSI/BSR) *Guide to Evaluation of Man-Machine Performance in Nuclear Power Generating Stations, Control Rooms, and Other Peripheries.*

BSR/IEEE 622B (Addenda to ANSI/IEEE 622-1987, for comment) *Recommended Practice for Testing and Startup Procedures for Electric Heat Tracing Systems for Power Generating Stations*, $26.00.

International Standards

This section includes publications for any of the three types of international standards:

—IEC standards (International Electrotechnical Commission)

—ISO standards (International Standards Organization)

—KTA standards [Kerntechnischer Ausschuss (Nuclear Technology Commission)]

Standards originating from the IEC and ISO can be obtained from the American National Standards Institute (ANSI), International Sales Department, 1430 Broadway, New York, NY 10018.

The KTA standards are developed and approved by the Nuclear Safety Standards Commission (NSSC) of the KTA, which is a component of the Gesellschaft für Reaktorsicherheit (Society for Reactor Safety) in Cologne, German Federal Republic. Copies of these standards can be ordered from Carl Heyman Verlag KG, Gereonstrasse 18-32, D-5000 Köln (Cologne) 1, German Federal Republic. These standards are in German, and their prices are shown in German currency (DM).

One IEC standard is included in this issue.

IEC

IEC 965:1989 (Published) *Supplementary Control Points for Reactor Shutdown Without Access to the Main Control Room*, $17.00.

Status of Power-Reactor Licensing Activities

Compiled by E. G. Silver

In this article, published as availability of new information warrants, we report on Nuclear Regulatory Commission (NRC) and industry activities related to the licensing of nuclear power-reactor projects in the United States. This includes all activities leading to the issuance of construction permits, zero-power and fuel-loading licenses, and low-power (i.e., up to 5% of full power) operating licenses for nuclear power reactors. Once a low-power license has been granted, the reactor in question is considered as a licensed facility, and further coverage relating to it is included in our regular article "Operating U.S. Power Reactors."

This article will also report on activities related to changes in the licensing process itself insofar as it applies to the types of licensing described previously. Thus, for example, activities related to the establishment of one-step licensing would be covered in this article, whereas actions related to operating license renewal will be covered under the rubric of "Operating U.S. Power Reactors." This issue covers information received during the first quarter of 1989.

Any NRC documents referenced in this article are generally available at the NRC Public Document Room, 1717 H Street NW, Washington, D.C. 20555.

Table 1 shows the status of all reactor projects in the licensing stage. This table is carried only when there are changes in it, but at least once a year in any event.

VOGTLE 2 RECEIVES LOW-POWER LICENSE

Georgia Power Company and the other owners of the Vogtle Nuclear Power Plant (Municipal Electric Authority of Georgia, Oglethorpe Power Corp., and the city of Dalton, Georgia) received a low-power license for Unit 2 authorizing fuel loading and testing up to 5% of full power on Feb. 9, 1989 (Ref. 1). Vogtle 2 is the second of two 1157 MW(e) PWRs in Burke County, Ga., on the bank of the Savannah River, about 26 miles south-southeast of Augusta, Ga., and 15 miles east-northeast of Waynesboro, Ga. Its sister unit, Vogtle 1, was licensed about two years earlier, in March 1987.

FINANCIAL MATTERS CONTINUE TO DOMINATE SEABROOK LICENSING ATTEMPTS

Amid indications that the seemingly never-ending efforts to license the long-completed Seabrook reactor may soon come to a successful conclusion, the financial strains that the long delay has caused continue to plague the Public Service Company of New Hampshire (PSNH), which was forced into bankruptcy by the financial strain mainly caused by the debt-service costs during the delay since the state did not allow any of the construction costs to be passed to ratepayers before the facility generated power.

At the start of the year, PSNH filed a reorganization plan in federal bankruptcy court in a maneuver designed to circumvent these restrictions and begin to recover its investment.[2] According to PSNH, a 30 to 40% rate increase would be needed to achieve financial health for the utility.

Under the plan, PSNH would establish a holding company regulated by the Federal Energy Regulatory Commission (FERC) which would allow the company to avoid the state law prohibiting utilities from charging customers for plants not in operation. The utility believes that FERC would be more sympathetic to such an increase than the state of New Hampshire.

The incoming governor of New Hampshire and the state's congressional delegation have stated that PSNH's plan is unacceptable because it would burden customers and curtail economic growth. Governor-elect J. Gregg said that he will introduce legislation into the New Hampshire senate to create a public power authority to take over the troubled utility. According to a Capitol Hill aide, this would not be the state's preferred solution, and it was seen by many primarily as a tool to make PSNH more amenable at the bargaining table.

C. Bayless, chief financial officer of PSNH, said that the announced plan was by no means

Table 1

NRC LICENSING STATUS OF POWER REACTORS IN THE PLANNING OR CONSTRUCTION STAGE

(As of the end of the first quarter of 1989)

REACTOR INFORMATION					CONSTRUCTION PERMIT			RECENT ACTIONS
			Proposed power level					
Name and location (owner/operator)	Reactor type (designer)	Docket No. (date)	MW(t)	MW(e)	ACRS action	Permit No. (date)	Projected fuel loading date	As of Mar. 31, 1989
BELLEFONTE 1 and 2, Scottsboro, Ala. (Tennessee Valley Authority)	PWR (B&W) PWR (B&W)	50-438 (6-73) 50-439 (6-73)	3600 3600	1213 1213	7-74 7-74	CPPR-122 (12-74) CPPR-123 (12-74)	Indefinite Indefinite	Inactive Inactive
CLINTON 2, Clinton, Ill. (Illinois Power Co.)	BWR (GE)	50-462 (10-73)	2894	933	4-75	CPPR-138 (2-76)	Unknown	NRC reviewing application for operating license
COMANCHE PEAK 1 and 2, Glen Rose, Tex. (Texas Utilities Generating Co.)	PWR (West) PWR (West)	50-445 (7-73) 50-446 (7-73)	3411 3411	1150 1150	10-74 10-74	CPPR-126 (12-74) CPPR-127 (12-74)	Unknown Unknown	NRC reviewing applications for operating licenses
FLOATING NUCLEAR PLANTS 1-8, Jacksonville, Fla. (Offshore Power Systems Inc.)	PWR (West)	STN 50-437 (7-73)	3411	1150	12-75	ML-1 (12-82)	Unknown	Inactive
GRAND GULF 2, Port Gibson, Miss. (Mississippi Power & Light Co.)	BWR (GE)	50-417 (11-72)	3833	1250	5-74	CPPR-119 (9-74)	Unknown	Inactive
LIMERICK 2, Pottstown, Pa. (Philadelphia Electric Co.)	BWR (GE)	50-353 (2-70)	3293	1065	8-71	CPPR-107 (6-74)	1990	NRC reviewing application for operating license
MIDLAND 1 and 2, Midland, Mich. (Consumers Power Co.)	PWR (B&W) PWR (B&W)	50-329 (1-69) 50-330 (1-69)	2452 2452	460 811	6-70 9-70	CPPR-81 (12-72) CPPR-82 (12-72)	Indefinite Indefinite	Construction halted
PERRY 2, Perry, Ohio (Cleveland Electric Illuminating Co.)	BWR (GE)	50-441 (6-73)	3579	1205	5-75	CPPR-149 (5-77)	Indefinite	Inactive
SEABROOK 1 and 2, Seabrook, N. H. (Public Service Co. of New Hampshire)	PWR (West) PWR (West)	50-443 (7-73) 50-444 (7-73)	3411 3411	1200 1200	12-74 12-74	CPPR-135 (7-76) CPPR-136 (7-76)	Loaded Indefinite	Unit 1 fuel loading and pre-critical testing licensed. Unit 2 inactive.
VOGTLE 2, Waynesboro, Ga. (Georgia Power Co.)	PWR (West)	50-425 (2-73)	3411	1113	4-74	CPPR-109 (6-74)	1989	Low-power license issued
WASHINGTON 1, Richland, Wash. (Washington Public Power Supply System)	PWR (B&W)	50-460 (10-73)	3600	1218	6-75	CPPR-134 (12-75)	Unknown	Inactive
WASHINGTON 3, Satsop, Wash. (Washington Public Power Supply System)	PWR (CE)	50-508 (8-74)	3800	1242	4-76	CPPR-154 (4-78)	Indefinite	Inactive
WATTS BAR 1 and 2, Spring City, Tenn. (Tennessee Valley Authority)	PWR (West) PWR (West)	50-390 (5-71) 50-391 (5-71)	3411 3411	1177 1177	9-72 9-72	CPPR-91 (1-73) CPPR-92 (1-73)	1992 Unknown	NRC reviewing applications for operating licenses

final and that the utility would prefer to remain under state authority if the state grants a "reasonable" rate increase. The state, however, favors 4% cost-of-living increases for the next 5 yr, which PSNH deems entirely insufficient.

The PSNH said that another option it was considering is the outright sale of the company. As the situation stood at the first of the year, if an agreement was not reached by February 1, other concerns, including the state of New Hampshire, would be free to bid on PSNH's assets. "If the state is serious, they will have to compete with other organizations putting up bids," a PSNH spokeswoman said. "They will have to come up with the best offer."

Aside from PSNH's $2 billion investment in Seabrook, the company reported that it was thriving financially, and it indicated that it had already received inquiries from 12 prospective buyers.

The company also emphasized again that it had not abandoned the idea of bringing Seabrook Station into commercial operation. Although the plant has been complete for several years, it has not gone on line because the state of Massachusetts has refused to file emergency evacuation plans for the Massachusetts communities that lie within the plant's emergency planning zone. The PSNH recently received a boost, however, when NRC indicated that it would not consider the company's bankruptcy when it reaches a final decision on the plant's low-power license.

The plan of PSNH was subject to approval by the bankruptcy judge hearing the case and other parties in the Chapter 11 filing. In addition, the utility's proposal to switch to federal regulation is subject to state approval and will set the stage for a precedent-setting legal showdown. The company said it will argue that the bankruptcy court can use its broad discretion in reorganizations to permit a switch to FERC over the state's objection.

Also in January 1989 the U.S. Supreme Court dismissed an appeal filed by PSNH challenging the New Hampshire state law that bars Seabrook from being included in the rate base because it has not yet produced electric power.[3] In handing down its order January 23, the court said it is dismissing the appeal "for want of a properly presented federal question."

The court's decision upheld a ruling by the New Hampshire Supreme Court Jan. 26, 1988, that denied higher rates to PSNH, which owns 35.6% of Seabrook through a consortium, saying that the utility was not entitled to a "bailout." The state court left intact a New Hampshire law barring rate increases to pay for a nuclear reactor until "the facility is used and useful in service to the public."

The PSNH challenged the statute after state regulators rejected the company's September 1987 request for a $71 million emergency rate increase plus other financial help to recover part of its $5.2 billion investment in Seabrook. The PSNH filed for bankruptcy protection from its creditors Jan. 28, 1988.

"The state statute is a reaffirmation of the 'used and useful' concept standing for the proposition that the direct consequences of investment risk for new plants must be borne by investors, not ratepayers," the state court ruled.

Public Service had argued that its investors were entitled to a "reasonable" return on their investment. In addition, the utility said that a rate increase would protect ratepayers as well by preventing the state from imposing on them the costly carrying charges that the New Hampshire law creates, "because it [would allow] the utility more gradual recovery on its total prudent investment."

The PSNH bankruptcy counsel R. Levin said that, although the company is "disappointed" with the court's order, it "should not have a major effect on Public Service's bankruptcy reorganization." He continued that the rate case was based on a need for an emergency rate increase to avert filing for bankruptcy.

In February 1989 PSNH agreed to pay $2 million to four former owners of Seabrook in return for an agreement not to sue over issues of claimed mismanagement of the project.[4] The agreement is part of a larger deal in which PSNH will assume $30 million in owed payments currently being ignored by the Massachusetts Municipal Wholesale Electric Company. Three other companies participating in the Seabrook project to a total of only 4%, however, have not agreed to the deal and may sue over the matter.

On Mar. 8, 1989, the new Governor of New Hampshire, J. Gregg, announced that that state was opposed to low-power operation of Seabrook (if and when such a license was issued) because, in his opinion, PSNH would not be financially sound enough to decommission the plant once it was radioactively contaminated.[5] Flying in the face of NRC licensing procedures, he averred that low-power testing should not begin until PSNH has achieved "all the clearances necessary to begin full-power operation."

Gregg said it was a desire to force PSNH to "exercise sound business judgment" that drove him to oppose low-power testing of the plant. The governor argued that low-power operation would contaminate the plant and "render the facility useless for alternate applications." He also suggested that a company in financial difficulty should not take on additional liability, as PSNH would do by posting $25 million as its share of a surety bond to assure the NRC of its ability to clean up the plant if a full-power license were denied.

Under former Governor J. Sununu, New Hampshire had been an unflinching supporter of Seabrook, and in later statements Gregg implied that the state's opposition to low-power testing would disappear if a rate increase agreement were reached with the company. The PSNH has been seeking increases of 30% over the next 3 yr, a figure the state considers much too high. The state proposed a 4% increase each year for 5 yr. That position was strengthened recently by a report commissioned by the New Hampshire Business and Industry Association which concluded that the state would lose as many as 22,600 manufacturing jobs if a 30% increase were imposed all at once. Job losses would be 15,200, the survey said, even with a one-time 10% increase.

In an interview Gregg said, "Our challenge to low-power testing is on the fiscal question. . . if we were able to reach agreement on rates. . . and it is a fair rate for a set period of time, it would moot the fiscal argument. It would take care of our challenge."

A spokesman for PSNH said the utility was concerned that the governor's decision could lead to further delays in providing needed power to New England. Another Seabrook partner, New Hampshire Yankee, issued a press release claiming that, although the governor had not indicated any opposition to the eventual operation of Seabrook, he was "misinformed" on issues regarding low-power testing. New Hampshire Yankee said that the "negligible" contamination during such testing would not render the facility useless and that in any case, "one does not wait until the last minute to perform low-power tests on a reactor."

The New Hampshire Yankee release also promised that the joint owners of Seabrook were committed to getting it on line and would shortly be filing the full $72.1 million surety bond with NRC.

Seabrook's owners did, in fact, buy such a bond on Mar. 20, 1989, which would guarantee up to $72 million for decommissioning the plant if it failed to be able to operate commercially after the tests.[6]

Another possible hindrance to licensing Seabrook was overcome when the Federal Emergency Management Agency certified that Maine's offsite emergency evacuation plan (a small piece of the state of Maine also falls within the Seabrook 10-mile radius) was "adequate to protect the health and safety of the public. . . [and was] capable of being implemented."

REFERENCES

1. NRC Press Release 89-25, Feb. 9, 1989.
2. *At. Energy Clearing House,* 35(1): 2 (Jan. 6, 1989).
3. *At. Energy Clearing House,* 35(4): 1 (Jan. 27, 1989).
4. *At. Energy Clearing House,* 35(6): 2 (Feb. 10, 1989).
5. *At. Energy Clearing House,* 35(10): 1 (Mar. 10, 1989).
6. *At. Energy Clearing House,* 35(12): 6 (Mar. 24, 1989).

Proposed Rule Changes as of Mar. 31, 1989[ab]

(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 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 1 10 CFR 2 10 CFR 9 10 CFR 73			12-30-88; 12-30-89	Reorganization of functions within the Office of Administration and Resources Management and minor corrective amendments	Final rule in 53:251 (52993); correction in 54:8 (1288)
10 CFR 2 10 CFR 20	12-2-86	3-2-87		Radioactive waste below regulatory concern; generic rulemaking	Advance notice of rulemaking published in 51:231 (43367)
10 CFR 2	2-4-87	4-6-87; 5-6-87	2-23-89; 3-27-89	Issuance of amendment; power following initial decision	Published for comment in 52:23 (3442); comment period extended in 52:68 (11475); final rule in 54:35 (7756)
10 CFR 2	5-29-87	7-28-87; 10-28-87	2-28-89; 3-30-89	Informal hearing procedures for materials licensing adjudications	Published for comment in 52:103 (20089); comment period extended in 52:142 (27821); final rule in 54:38 (8269)
10 CFR 2 10 CFR 51 10 CFR 60	5-5-88	8-3-88		NEPA review procedures for geologic repositories for high-level waste	Published for comment in 53:87 (16131)
10 CFR 2	11-3-88	12-5-88		Rule on submission and management of records and documents related to the licensing of a geologic repository for the disposal of high-level radioactive waste	Published for comment in 53:213 (44411)
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); correction in 54:51 (11224)
10 CFR 9			3-10-89; 3-10-89	Freedom of Information Act; appeal authority for Deputy Executive Director	Final rule in 54:46 (10138)

Proposed Rule Changes as of Mar. 31, 1989 (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 15	10-7-88	11-21-88		Debt collection procedures	Published for comment in 53:195 (39480)
10 CFR 19	11-14-88	1-10-89; 2-9-89		Sequestration of witnesses interviewed under subpoena	Published for comment in 53:219 (45768); comment period extended in 54:4 (427)
10 CFR 20	8-29-88	10-28-88		Disposal of waste oil by incineration	Proposed rule in 53:167 (32914)
10 CFR 21 10 CFR 50	11-4-88	1-3-89		Criteria and procedures for the reporting of defects	Published for comment in 53:214 (44594)
10 CFR 26	9-22-88	11-21-88		Fitness-for-Duty Program	Published for comment in 53:184 (36795)
10 CFR 30 10 CFR 40 10 CFR 70	4-20-87	7-20-87		Emergency preparedness for fuel cycle and other radioactive material licensees	Published for comment in 52:75 (12921)
10 CFR 34	3-15-88	5-16-88; 8-16-88		Safety requirements for industrial radiographic equipment	Published for comment in 53:50 (8460); comment period extended in 53:98 (18096)
10 CFR 35	10-2-87	12-1-87		Basic quality assurance in radiation therapy	Published for comment in 52:191 (36942)
10 CFR 35	10-2-87	12-31-87		Comprehensive quality assurance in medical use and a standard of care	Published for comment in 52:191 (36949)
10 CFR 35	5-25-88	8-24-88		Medical use of byproduct material; training and experience criteria	Advanced notice of proposed rulemaking in 50:101 (18845)
10 CFR 40	8-25-88	10-24-88		Custody and long-term care of uranium mill tailings sites	Advanced notice of proposed rulemaking in 53:165 (32396)
10 CFR 50	10-29-86	1-26-87; 4-24-87		Leakage rate testing of containments of light-water-cooled nuclear power plants	Published for comment in 51:209 (39538); comment period extended in 52:14 (2416)

(Table continues on the next page.)

Proposed Rule Changes as of Mar. 31, 1989 (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	11-6-86	1-5-87; 2-2-87		Production and utilization facilities; request for comments on development of policy for nuclear power plant license renewal	Policy statement published for comment in 51:215 (40334); comment period extended in 51:250 (47249)
10 CFR 50	11-27-87	1-25-88		Integrated schedules for implementation of plant modifications; proposed policy statement	Published for comment in 52:228 (45344)
10 CFR 50 10 CFR 73	3-9-88	5-9-88		Nuclear power plant access authorization programs; policy statement	Published for comment in 53:46 (7534)
10 CFR 50	4-6-88	7-5-88; 8-5-88		Leak-Before-Break technology; solicitation of public comments on additional applications	Published to solicit comments in 53:66 (11311); comment period extended in 53:134 (26447)
10 CFR 50	6-13-88	7-13-88	2-22-89; 2-22-89	Cooperation with states at commercial nuclear power plants and other nuclear production or utilization facilities; policy statement	Published for comment in 53:113 (21981); final rule in 54:34 (7531)
10 CFR 50	7-19-88	8-18-88	2-17-89; 3-20-89	Licensee action during national security emergency	Published for comment in 53:138 (27174); final rule in 54:32 (7179)
10 CFR 50	8-29-88	10-28-88		Nuclear plant license renewal	Advanced notice of proposed rulemaking in 53:167 (32919)
10 CFR 50	9-19-88	10-19-88	3-17-89; 3-17-89	Extension of time for the implementation of the decontamination priority and trusteeship provisions of property insurance requirements	Published for comment in 53:181 (36338); final rule in 54:51 (11161)
10 CFR 50	10-24-88	12-23-88		Flow control conditions for the standby liquid control system in boiling water reactors	Published for comment in 53:205 (41607)
10 CFR 50	11-28-88	1-27-89; 2-27-89		Ensuring the effectiveness of maintenance programs for nuclear power plants	Published for comment in 53:228 (47822); comment period extended in 53:250 (52716)

Proposed Rule Changes as of Mar. 31, 1989 (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 10 CFR 55			1-24-89; 1-24-89	Policy statement on the conduct of nuclear power plant operations	Final rule in 54:14 (3424)
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 52	8-23-88	10-24-88; 11-7-88		Early site permits; standard design certifications; and combined licenses for nuclear power reactors	Published for comment in 53:163 (32060); comment period extended in 53:205 (41609)
10 CFR 55	12-29-88	2-27-89; 3-29-89		Education and experience requirements for senior reactor operators and supervisors at nuclear power plants	Published for comment in 53:250 (52716); comment period extended in 54:37 (8201)
10 CFR 60	2-27-87; 5-18-88	4-29-87; 6-29-87; 7-18-88		Definition of "high-level" radioactive waste	Advanced notice published for comment in 52:39 (5992); comment period extended in 52:86 (16403); published for comment in 53:96 (17709)
10 CFR 62	12-15-87	2-12-88	2-3-89; 3-6-89	Criteria and procedures for emergency access to non-Federal and regional low-level waste disposal facilities	Notice of intent to develop regulations in 52:10 (1634); published for comment in 52:240 (47578); correction in 53:15 (1926); final rule in 54:22 (5409)
10 CFR 70 10 CFR 74			2-15-89; 2-15-89	Centralization of material control and accounting licensing and inspection activities for non-reactor facilities	Final rule in 54:30 (6876)
10 CFR 71	6-8-88	10-6-88; 12-6-88; 3-6-89		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)

(Table continues on the next page.)

Proposed Rule Changes as of Mar. 31, 1989 (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 76	4-22-88	7-21-88; 10-22-88		Regulation of uranium enrichment facilities	Published for comment in 53:78 (13276); comment period extended in 53:179 (35827)
10 CFR 140	12-20-88	1-4-89		Financial protection requirements and indemnity agreements; miscellaneous amendments necessitated by changes in the Price-Anderson Act	Published for comment in 53:244 (51120)
10 CFR 150	8-22-88	10-21-88		Reasserting NRC's authority for approving on-site low-level waste disposal in Agreement States	Published for comment in 53:162 (31880)
10 CFR 170 10 CFR 171	6-27-88	7-27-88	8-12-88; 9-12-88; 12-29-88; 1-30-89	Revision of fee schedule	Published for comment in 53:123 (24077); interim rule in 53:156 (30423); final rule in 53:250 (52632); corrections in 54:14 (3558)

[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.

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Book Review of Nuclear Accidents: Intervention Levels for the Protection of the Public

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Living PRA Computer Systems

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Summary of ICAP Assessments of RELAP5/MOD2

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Thermal Performance Monitoring System at Maanshan Nuclear Power Plant

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Warning Systems for Nuclear Power Plant Emergencies

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Steam Generator Tube Performance: Experience with Water-Cooled Nuclear Power Reactors During 1985

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Systems Interaction Analyses: Concepts and Techniques (Part II)

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INTERNATIONAL TOPICAL MEETING ON THE SAFETY, STATUS, AND FUTURE OF NON-COMMERCIAL REACTORS AND IRRADIATION FACILITIES

Boise, Idaho, Sept. 30–Oct. 4, 1990

Call for Papers

This conference is sponsored by the Idaho Section and the Nuclear Reactor Safety Division of the American Nuclear Society (ANS) with cosponsorship from the Atomic Energy Society of Japan and the Commission of the European Communities.

Papers are solicited on all research and engineering aspects of the safety, design, operating experience, risk assessment, policy, and teaching and training relating to noncommercial reactors and irradiation facilities. Sessions are planned on the following topics: *Safety Aspects of Plant Design Characteristics*—Fuel and Core Design, Nuclear Island Design, Engineering Safety Features, Containment/Confinement, Plant Control and Automated Assistance to Operator, Passive Safety Design, New Reactor Concepts, NPR Design Concepts, Design Goals: Accident Prevention vs. Mitigation. *Operating Experience/Incident Experience*—Plant Aging/Plant Life Extension, Safety Enhancing Operation and Maintenance Practices, Safety Related Human Factors, Facility Modification to Enhance Safety, Results of Safety Reviews, Improved Technical Specifications, Improved Operating/Emergency Procedures, Emergency Planning and Preparedness, Safety Related Significant Events and Accidents. *Safety Analysis and Risk Assessment*—Design Basis Accident Research, Severe Accident Research, Computer Code Development and Validation, Loss of Cooling Studies, Reactivity Insertion Accident Studies, Safety Assessment of Mature Reactors, PRA Experience and Innovative Techniques. *Safety Policy for Current and Future Reactors*—Application of Commercial Power Plant Experience and Safety Criteria to Non-Commercial Reactors, Licensing Approaches/Issues, Safety Goals, Design/Backfit Criteria, Safety Oversight and Regulation Criteria. *Teaching and Training Applications*—Simulator Design and Use in Training, Advanced Training Techniques, Training Criteria and Testing, Simulator Based Training Program, Use of Simulators in Validating Human Factors Models.

The deadline for submission of summaries is Jan. 15, 1990. Full papers of accepted contributions will be due Aug. 1, 1990.

To submit papers or obtain more information, contact Mr. Doug Croucher, Program Chairman, EG&G Idaho, Inc., P.O. Box 51218, Idaho Falls, ID 83401-1218. Telephone: (208) 526-9804.

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