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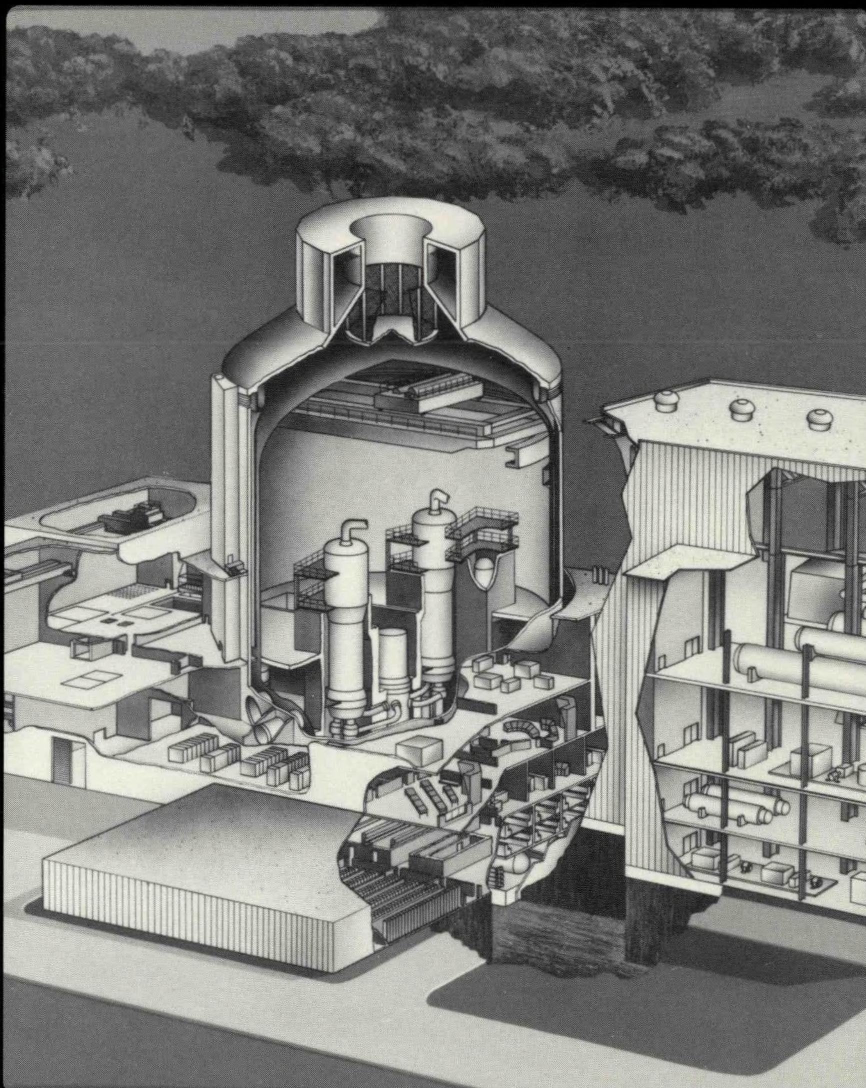

TECHNICAL PROGRESS JOURNAL JAN • MAR 1992

The Operational Performance Technology Section (OPT) of the Engineering Technology Division of the Oak Ridge National Laboratory is pleased to announce the availability of the *Licensee Event Report (LER) Compilation* for February 1992, a report heretofore prepared monthly by OPT for the Office for Analysis and Evaluation of Operational Data of the NRC. Each monthly report summarizes all LERs submitted during the previous monthly reporting period. The summaries are arranged alphabetically by facility, and chronologically by event date for each facility. Component, system, and keyword indexes are included. These monthly reports have heretofore been available from the NRC/GPO Sales Program, Superintendent of Documents, Government Printing Office, Washington, DC 20402. However, the series has been discontinued, and the issue containing the March 1992 reports will be the last one. The collection of the events and their entry into the Sequence Coding and Search System (SCSS) data base will continue. Single copies of the publication are still available from the National Technical Information Service, Springfield, VA 22161.

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Vol. 11, No. 2

Licensee Event Report (LER) Compilation

For month of February 1992

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

The Nuclear Operations Analysis Center

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

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

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

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

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

Cover: Our cover picture this month is an artist's cut-away rendering of a Westinghouse Advanced Passive 600 Plant. This plant is described in this issue in an article by McIntyre and Beck. The air cooling vent and water storage tanks at the top of the containment building are clearly visible, as is the two-loop design layout.

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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 aim at 20 double-spaced typed pages (including *figures, tables, and references*). *Send inquiries* or 3 copies of manuscripts (with the draftsman's original line drawings plus 2 copies and with black-and-white glossy prints of photographs plus 2 copies) to E. G. Silver, Oak Ridge National Laboratory, P. O. Box 2009, Oak Ridge, TN 37831-8065.

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

Editorial Staff

General Safety Considerations

Edited by G. T. Mays

Technical Note: A New Approach to Fission Reactor Safety

By Yu. V. Petrov[a]

Abstract: *For the development of safe nuclear energy production acceptable to society, the future use of strongly subcritical reactors (k = 0.96–0.97) driven by proton or deuteron accelerators is proposed. An accelerator with a current of 40 mA and particle energy of about 0.8 GeV/nucleon will provide 2 GW(t) reactor power. This article discusses the design, control, and parameters of such a system.*

The triumphant growth of atomic energy production was recently halted by the Three Mile Island accident in 1979 and the Chernobyl catastrophe in 1986. Measurable parts of the community are now protesting against the wide use of atomic energy; it is considered too dangerous. The competitiveness of atomic energy is not now a sufficient argument to build new nuclear power stations, and the number of contracts for their construction has decreased drastically all over the world. This means that the first stage of development of the atomic energy industry is finished; the second period has begun. The future of atomic energy will be determined mainly by whether or not the public considers it sufficiently safe. In Dr. A. Weinberg's words, reactors must be not only inherently but also "transparently" safe.[1]

Normally operating nuclear power plants produce fewer environmental effects than coal and oil power plants. The complete nuclear fuel cycle, which includes mining, uranium enrichment, fuel-element fabrication, and production of electricity, is much less harmful to people and the environment than fossil-fuel energy production per unit of produced electric energy. Even many opponents of atomic energy agree with this. The main public fear is the possibility of an uncontrolled accident leading to a global catastrophe on the Chernobyl scale. In this situation, one possibility is to throw away a cornerstone of the physics of fission reactors—the concept of their criticality—and switch to strongly subcritical systems.[2]

ACCELERATOR-DRIVEN NUCLEAR FISSION REACTORS

It is well-known that a key role in the control of power growth in a nuclear reactor is played by delayed neutrons. The fraction β of these neutrons is very small: β = 0.3–0.7%. Positive reactivity excess should be much less than this ($\rho << \beta$). If, on the other hand, ρ is greater than β, the reactor is prompt critical and its power, P, satisfies the equation

$$\frac{dP}{dt} = \frac{1}{\ell}\left(\rho_{eff} P + \dot{P}_s\right) \tag{1}$$

where $\rho_{eff} = \rho - \beta$ is greater than zero, ℓ is the neutron lifetime, and P_s is the power of the external source, which is usually negligible. If ρ_{eff} is of the order of 0.01, the power grows rapidly and exponentially with a time scale of $\tau = l/\rho_{eff}$ between 10^{-2} and 10^{-1} s, and the system

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of control rods is not fast enough to stop it. This was exactly the case in Chernobyl Unit 4, in which the void coefficient turned positive, several times as large as β (Ref. 3). If this reactor had been subcritical, such a catastrophe would have been impossible. The critical state of the reactor ($\rho = 0$) is too close to the dangerous limit of $\rho = \beta\,(\rho_{eff} = 0)$.

Nevertheless, a strongly subcritical power reactor is possible. If the reactivity of a reactor is negative $\rho_{eff} = \rho = -|\rho|$, then, in the steady state ($dP/dt = 0$), we have from Eq. 1

$$P = \frac{P_s}{|\rho|} = \frac{Sk}{1-k} \quad (k<1) \qquad (2)$$

An intense neutron source is needed for a reactor to have a high enough thermal power. Such a source was proposed as early as at the end of the 1940s by N. Semenov in the U.S.S.R. and by E. Lawrence in the United States and somewhat later by W. Lewis in Canada. The idea was as follows: A beam of protons with energies T_0 between 0.6 and 1.0 GeV impinges upon a heavy isotope target (Pb, Bi, U). Because of the nucleus–nuclei collisions, spallation reactions occur. Highly excited nuclei emit several neutrons. Their energy is sufficient to produce fission of the most abundant isotope of uranium, ^{238}U, surrounding the target.

Neutrons with energies less than the fission threshold are captured by ^{238}U and produce plutonium (^{239}Pu). This is known as the electronuclear (EN) method of energy and plutonium production.

The number of fissions per unit energy of fast nucleons of the beam is related to ρ by[4]

$$Z = Z_h + \frac{Z_0}{|\rho|}, \quad Z_0 = \frac{S_h}{\bar{\nu}_f} \qquad (3)$$

Here Z_h is the number of fissions as the result of primary fast nucleons, S_h is the number of produced fast neutrons, and $\bar{\nu}_f$ is the number of secondary fission neutrons averaged over the energy spectrum and fissile isotopes. For $k < 1$ and $|\rho| << 1$, the value of $1/|\rho|$ may be large enough to make accelerator power acceptable.

DEGREE OF REACTOR SUBCRITICALITY

For reactor safety, the degree of subcriticality must be as high as possible. On the other hand, economic restrictions require the recirculation power fraction α to be no higher than 15%.

Neglecting Z_h in Eq. 3, we have

$$\alpha = \frac{1}{K_B} = \frac{P_a}{P_e}, \quad |\rho| = Z_0 E_f \eta_a \eta_e \alpha \qquad (Z_h E_f \eta_a \eta_e \alpha << 1) \qquad (4)$$

Here K_B is the beam energy gain factor—the ratio of produced electric power P_e to power P_a consumed in operating the accelerator, η_a is the accelerator efficiency, and η_e is the efficiency of transformation of heat to electricity. The value of Z_0 depends on target and blanket structure, their sizes, etc. For example, in the case of a gaseous DT-target with a blanket of metallic uranium (U_{nat}; 54% vol%), structure materials (Fe; 12%) and helium coolant (FGM-blanket) calculations give $Z_0 \simeq$ 10 fissions/GeV for a deuteron beam with energy T_0 = 0.8 GeV/nucleon.[5] As to α, it is desirable to leave its value as low as is permitted by the subcriticality decrease.

Time variations of reactivity depend on reactor type. The largest excess of reactivity reserved for the power coefficient, for steady-state poisoning by Xe and Sm, and for fuel burnup compensation is required for light-water reactors with oxide fuel (Fig. 1) (Ref. 6). These effects are half as great for fast reactors with oxide fuel. It is remarkable that, for fast reactors with sodium coolant and metallic U–Pu fuel, the total change of reactivity excess may be very small: $|\Delta\rho| = 2\beta$ (<1%). For the reactor to remain strongly subcritical during such changes, it is necessary that $|\rho|$ be several times larger than $|\Delta\rho|$.

Another possible origin of changes in ρ may be a loss of coolant or a shape perturbation of the reactor as the result of an accident. The danger of coolant loss is minimal for gas-cooled reactors with metallic fuel.[5,7] As for the changes of shape, for cylindrical reactor perturbations, $\Delta\rho$ will be negative and of the second order in relative deformation.[8] Therefore, to guarantee that criticality (ρ = 0) will never be reached, one may safely assume that $\rho = -0.03$. Then we have $\alpha = 0.075$, $\eta_{eN} = \eta_e (1-\alpha) = 0.37$, and for P_e = 800 MW(e), the accelerator will have acceptable power demands P_a = 60 MW(e).

For η_a = 0.5, the power of the accelerator beam will be 30 MW. Meson factories in operation have proton energies of 600 to 800 MeV and beam power of ~1 MW (LAMPF in Los Alamos, United States, and the PSI accelerator in Villigen, Switzerland). Several accelerators designed recently have beam powers of hundreds of megawatts and $\eta_a \simeq$ 0.5–0.6 (see, e.g., Ref. 9). The author of Ref. 9 estimated the total accelerator cost to be

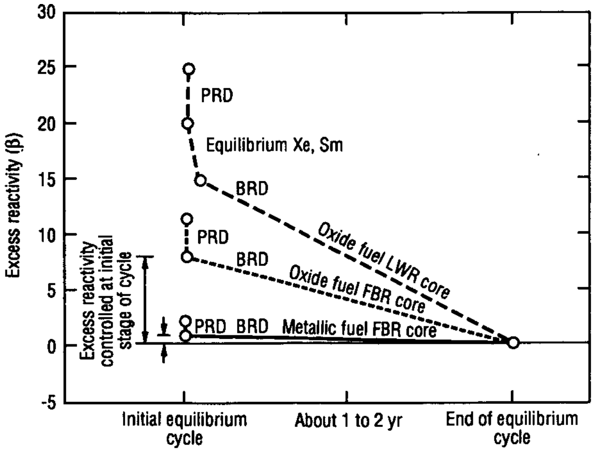

Fig. 1 Excess reactivity of oxide and metallic fueled core[6] (initial cycle and time variation). PRD, power reactivity decrement; BRD, burnup reactivity decrement; FBR, fast breeder reactor.

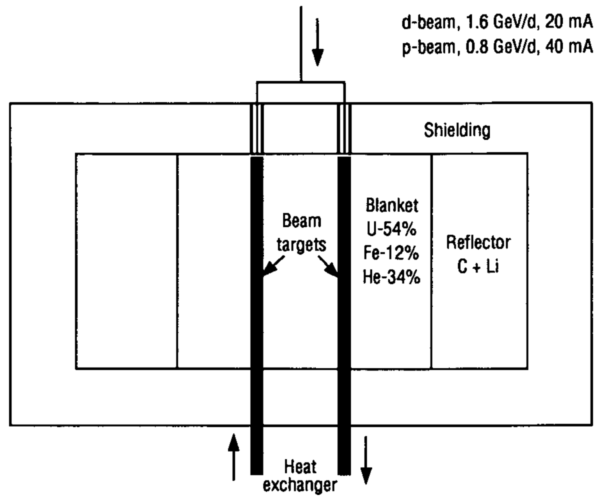

Fig. 2 Elevation view of subcritical reactor driven by an accelerator. d, deuteron; p, proton.

about \$400 million. The possibility of their wide use as reactor drivers will intensify realization of these projects.

REACTOR DESIGN

It is possible to consider several types of subcritical reactors. The traditional way is to use U–Pu fuel, but the use of Th–U fuel can reduce the amount of such radioactive wastes as long-lived actinides.

A beam of fast protons (or deuterons) must be split into several beams, each of them impinging on a separate target (Fig. 2). The target may consist of heavy or light elements, such as a dense gas, liquid lithium, and water (D_2O or H_2O). In the last case a considerable fraction of the fast neutrons of the beam is scattered into the blanket, which transfers 70% of the initial energy.[5,10] Because of the low density of the target, it is possible to have a free path length of the nucleons of the beam comparable with the height of the reactor. This results in a more uniformly scattered nucleon flux on the blanket wall. The internal heat released in the target can easily be removed by using the target itself as a coolant.

The additional volume needed for the beam targets becomes available since control rods become unnecessary. The volume of the rods in RBMK-type reactors is more than 10%. Power control in a subcritical, driven reactor will be provided by a feedback signal from the reactor (produced by measuring the reactor's neutron power, temperature, etc.) to the accelerator. This allows stabilization of system power because the system is capable of reacting quickly to any fluctuations of the reactor's parameters. The time scale of power changes in a metallic U–Pu-fueled reactor is $\tau = l/\,|\rho+\beta| \simeq 10^{-5}$ s. Hence there is a gain in reaction time of several orders of magnitude in comparison with what can be achieved by control rods.

For the conventional EN systems discussed earlier in the literature (with $k = 0.4$–0.5), there are difficulties with heat removal arising from nonuniformity of energy release near the source. For a subcritical reactor with $|\rho| << 1$, these problems become much less serious since the typical relaxation length of power release is $L = L_0/(1 - k)^{1/2}$, where L_0 is the neutron migration length in a nonmultiplying system. If the targets are separated by distances of $a \simeq 2L \simeq 10\ L_0$, a reasonably flat power distribution will result.

For $k = 0.96$–0.97, the mean concentration of ^{239}Pu in the uranium should be 6% for the FGM blanket. The plutonium concentration in the fuel elements must be in the range of 0 to 12% for a reactor operating in a stationary mode. Some of the now-available weapons plutonium could be used as the first charge. Thereafter the reactor will be loaded with fresh natural uranium assemblies. After irradiation the fuel elements will have plutonium concentrations of about 12%.

Burnup of natural uranium in unloaded fuel elements will be about 8 to 9%. Compared with light-water reactors, the amount of primary natural uranium needed for electricity production will be about 10 times less. This means that all mining, enrichment, and transportation efforts will also decrease by an order of magnitude. The same is true for the total environmental impact. The amount of radioactive waste per gigawatt-year (electric) will remain the same. Uranium enrichment becomes unnecessary. Spent fuel elements will first be stored and then buried (a once-through fuel cycle). If some plutonium is needed to start a new reactor, it can be extracted from the spent fuel.

Along with fast nucleons, neutrons arising from muon catalyzed DT-fusion can also be used: $d + t \rightarrow {}^4He + \eta$ (Ref. 7). In an experiment $X_c \sim 150$ fusions per one muon were detected in a dense DT-mixture,[11] which is in excellent agreement with tentative theoretical predictions.[12] In the blanket, fast fusion neutrons cause uranium fission and build up plutonium. Calculations show that the mesocatalytic (MC) method of energy and neutron production has parameters close to those of the EN method. The combination of both methods in one device would double the energy release in the blanket and the plutonium yield.[7] Correspondingly, half the accelerator power would then be needed per unit of energy produced. The blanket and target first-wall fluxes will also be halved. Muon confinement would be provided by a rather large constant magnetic field (about 10 T). Such fields are used in designs of modern thermonuclear reactors. Tritium for the DT-reaction can be produced from lithium placed in a graphite or heavy-water moderator behind the uranium zone. It is difficult to judge now whether the gain from using the MC process would justify the complication of the system. The intense current research in the field of muon catalysis phenomena may soon be expected to provide the answer.[13]

CONCLUSIONS

Safe nuclear energy that is acceptable to the public requires a search for nontraditional approaches. One of the possibilities is rejecting reactor criticality. Strongly subcritical reactors ($k = 0.94$ to 0.97) driven by an accelerator are capable of producing electric power on the level of a modern nuclear power plant. For example, helium-cooled fast subcritical reactors ($k = 0.97$) with a metallic fuel can have a power of 800 MW(e) provided that the current of the proton beam ($T_0 \simeq 0.8$ GeV) is about 40 mA. Control of such a reactor is much easier and safer than for a critical one. Only the power of the accelerator needs to be controlled. The slow complicated system of control rods needed in critical reactors would no longer be needed. If, along with the fast nucleons, mesocatalytic neutrons of a DT-fusion reaction were used, the parameters of the system would be considerably improved. The cost of doing so would be a more complicated design.

The accelerator-driven subcritical assembly (ADSCA) concept totally eliminates the possibility of a reactor runaway and a Chernobyl-type catastrophe. The possibility of a loss-of-coolant accident (analogous to the one at Three Mile Island) would still exist. Since criticality, however, would no longer be required, there would be more room to introduce inherently safe means to handle the decay heat.[a] As proposed by A. Sakharov and E. Teller (and also by K. von Weizsäcker), the ADSCA would be naturally situated underground. This would strongly diminish the impact of radioactivity release in an intrusive accident. The underground location would also make the reactor safe in case of common or civil war. Two plutonium-producing reactors and one power plant near Krasnojarsk (Russia) are now sited deep underground, below a 250-m-thick layer of rock.[14]

The idea of using an accelerator in combination with a subcritical blanket is not new. G. N. Flerov and I. I. Gurevitch made a similar proposal as early as 1946 when the problem of achieving criticality still was not completely solved. Later, in 1965, C. Millar proposed the use of such a system with ($k = 0.99$) as a plutonium breeder.[15] At that time problems of nuclear safety were much less acute, and the same reactor without the accelerator ($k = 1$) seemed simpler and more attractive.[16] However, time has changed the priorities.

When this study was nearly complete, I learned of the work by Dr. H. Takahashi, in which he proposed to use a subcritical reactor (with $k = 0.99$) driven by an accelerator for transuranium nuclide incineration.[17] Another project was proposed recently by a group at Los Alamos National Laboratory using a system with $k = 0.8$ to 0.9 (Refs. 18 and 19). The authors intended to transmute the most dangerous long-lived fission products (e.g., ^{137}Cs, $T_{1/2} = 30$ years). However, because of the very small neutron capture cross section ($\sigma_a \sim 0.1b$) for ^{137}Cs, the neutron flux for thermal and resonance energies would need to be greater than 10^{16} neutrons cm^{-2} s^{-1}. To achieve such an enormously high neutron flux and simultaneously maintain high transmutation efficiency in the neutron balance is a rather complicated problem. At present it seems to be more important to discuss all global aspects of the ADSCA concept than to concentrate on any specific project.

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I am indebted to Dr. V. Kuz'minov for his help in calculations and to Dr. H. Takahashi who made his report[17] available to me. The discussions with

[a]The author is indebted to Dr. A. Weinberg for this remark.

Dr. Ch. Bowman were very helpful. I gratefully acknowledge the support of the main ideas of this proposal by Profs. T. D. Lee, K. Siegban, E. Teller, E. Velikhov, A. Zichichi, and other participants of the seminar "Planetary Emergencies" in Erice (August 1991). I wish especially to thank Dr. A. Weinberg for his valuable remarks.

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ERRATUM

In *Nuclear Safety*, Vol. 32, No. 4, in the article "Good Relationships are Pivotal in Nuclear Data Bases," by A. S. Heger and B. V. Koen, a Department of Energy report (page 491, first text paragraph) was mentioned but was not referenced.

The reference should have been given as:

B. Tashjian et al., *An Analysis of the Reliability of Light Water Reactor Power-Actuated Pressure-Relieving Valves and Safety (Relief) Valves and Their Components Using the Nuclear Plant Reliability Data System (NPRDS)*, Report SWRI-17-6649, NTIS.

Accident Analysis

Edited by R. P. Taleyarkhan

Analysis and Modeling of Fission Product Release from Various Uranium–Aluminum Plate-Type Reactor Fuels

By R. P. Taleyarkhan[a]

Abstract: *This article provides a perspective overview and analysis of volatile fission-product release data obtained for uranium–aluminum (U–Al) reactor fuels, U–Al$_x$ (alloy and dispersed), U_3O_8–Al (dispersed), U_3Si_2–Al (dispersed), and U_3Si–Al (dispersed). Several shortcomings in the experimental data bases have been highlighted and areas of uncertainty related to extrapolation of correlation predictions identified. Fission-product release characteristics for the U–Al reactor fuels have been shown to be radically different from those for the light-water-reactor fuels. Significant dependencies that exist on ambient medium, rate of release, fuel type and structure, fission-product chemistry, and burnup are analyzed. The potential inaccuracies involved in using the CORSOR (i.e., first-order exponential) approach for modeling release rate dependence for U–Al fuels with time are demonstrated, and an alternate formulation that captures characteristic trends is suggested. A library of correlations and methods for predicting release rates (for each individual volatile species) that may vary with time, ambient environment, burnup, and temperature were developed on the basis of analysis of characteristic data trends, phenomenological aspects, and regression analysis. A comparison of suggested correlation predictions against measurements resulted in an overall mean value and standard deviation of 1.02 and 0.22, respectively, over the entire temperature range investigated. Overall, statistics improve significantly (mean, 1.0; standard deviation, 0.09) if the Hanford Engineering Development Laboratory data are excluded.*

It is well known that radioactive fission products generated in nuclear fuel constitute the principal hazard to the general public associated with severe accidents. Because of this and the results of the well-known *Reactor Safety Study*,[1] considerable experimental and analytical work has been conducted over the past decade to evaluate fission-product release from light-water-reactor (LWR) fuels and to establish models for inclusion in computer codes for severe accident analyses. Only limited analyses, however, have been performed on the experimental data obtained so far for uranium–aluminum (U–Al) reactor fuels. The purpose of this article is to (1) provide a perspective overview and analysis of data obtained for the U–Al reactor fuels U–Al$_x$, U_3O_8–Al, U_3Si–Al, and U_3Si_2–Al; (2) present mathematical formulations for the change in time of release rates of fission products from heated U–Al fuels in various environments; and (3) present a library of correlations for such fuels on the basis of analysis of characteristic data trends and regression analysis.

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FISSION-PRODUCT RELEASE EXPERIMENTS FOR U-Al-FUELED REACTORS

Systematic data for volatile fission-product release from U–Al fuels come principally from four sources: (1) noble gas release data for U–Al$_x$ dispersion fuel,[2] (2) fission-product release data obtained for U–Al alloy fuels,[3] (3) fission-product release data obtained for U–Al alloy and U_3O_8–Al cermet Savannah River Production reactor fuels,[4] and (4) fission-product release data for dispersed U–Al$_x$, U_3Si_2–Al, and U_xSi_y–Al fuels by Saito et al.[5,6] at the Japan Atomic Energy Research Institute

(JAERI). Salient aspects of these experimental programs are given in Table 1.

As noted in Table 1, data were obtained in a variety of ambient media at the Oak Ridge National Laboratory (ORNL) and Hanford Engineering Development Laboratory (HEDL). The JAERI data, however, were obtained only in air. The experiments were conducted with specimens of about 24 and 65 isotopic (^{235}U) percent burnup. The HEDL experiments were conducted with specimens of about 52 isotopic (^{235}U) percent burnup, whereas the Shibata et al.[2] experiments were conducted with fuel specimens that had a burnup of about 62 isotopic (^{235}U) percent. A third major distinction deals with the heating time of fuel specimens at various temperatures, which were either 120 or 3600 seconds. Most of the ORNL and HEDL data were collected with fuels heated at a given temperature for 2 minutes, whereas some of the ORNL data and all the JAERI data were collected over 60 minutes.

ANALYSIS OF FISSION-PRODUCT RELEASE DATA

The main purpose of the data obtained by Shibata et al.[2] was to measure the threshold temperature at which a significant release of noble gas occurs. These were low-temperature tests where fuel temperatures were limited to 973 K. The release of fission products from the fuel plates at temperatures below 773 K was negligible (i.e., < 0.1%). At higher temperatures, noble gas release occurred in three stages. The first rapid release was observed upon initiation of fuel blistering (833 K). The next release coincided with the solidus temperature of Al-6061 alloy at 858 K. Finally, the last stage occurred at about 923 K, which corresponded with the eutectic temperature of the U–Al alloy. Negligible amounts (i.e., < 0.1%) of iodine and cesium were released in these tests. Noble gas release from U–Al alloy, U_3O_8–Al, and dispersed U–Al fuel specimens (in the JAERI experiments) was found to be almost identical to that seen by Shibata et al. As can be seen from Fig. 1, however, noble gas release from silicide fuels was found to be radically different from that for the other fuels. Significant burnup dependence is seen, which indicates substantial retention capability in the dispersed uranium silicide grain matrix.

As a cautionary note, it should be recognized that "negligible" releases of fission gases below certain temperatures presupposes application of these results to severe accidents in nuclear reactors (the intended application of this article). Negligible to a severe accident researcher may be unacceptably large for fuel-handling purposes.

Measurements of other volatile fission-product species (viz., cesium, iodine, and tellurium) and also ruthenium (which is normally considered nonvolatile) from the other three data bases revealed significant dependencies on the ambient medium, fuel burnup, heating–collection time, and fuel composition. It was generally found that (1) oxidizing environments greatly enhance the release of iodine, cesium, and tellurium; (2) release amounts generally increased with fuel burnup; (3) the rate of release varied

Table 1 Salient Aspects of Fission-Product Release Experimental Programs[a]

Institution (researchers)	Fuel type	Burnup, %	Enrichment, %	Ambient	Temperature range, K	Heating time, min	Principal species investigated
HEDL (Woodley et al.)[4]	UAl_4, U_3O_8	52	60 to 80	Air, steam Argon	973 to 1373	2.5	Noble gases, I, Cs, Te
ORNL							
(Parker et al.)[3]	UAl_4	24	93	Air, steam Helium	973 to 1373	2, 60	Noble gases, I, Cs, Te, Ru
(Shibata et al.)[2]	Dispersed UAl_x	62	40	Helium	<973	30	Noble gases
JAERI (Saito et al.)[5]	(Fuel dispersed in Al) U_3Si_2–Al UAl U_3Si_2–U_3Si–Al	23 and 65	20	Air	973 to 1373	60	Noble gases, I, Cs, Te, Ru

[a]Fuel burnup is related to the ^{235}U isotope. Fuel enrichment levels for the ORNL and HEDL tests were not documented and are best guesses on the basis of personal conversations.

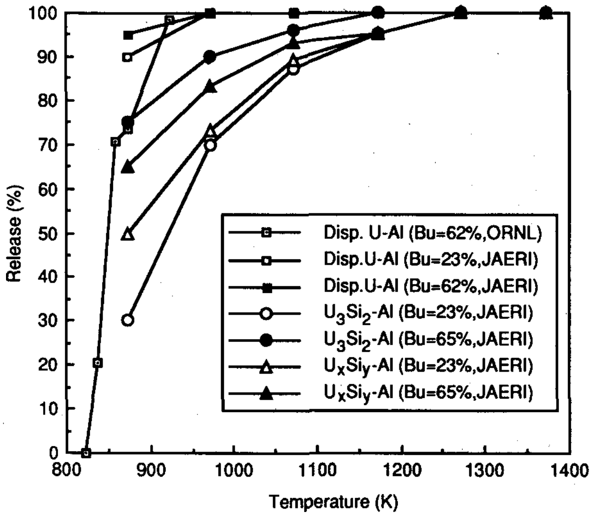


Fig. 1 Noble gas release variation with U–Al fuel type and burnup (time at temperature, 3600 seconds).

substantially with time and temperature; (4) significant fission-product retention in the fuel matrix was observed at temperatures well above the boiling points of individual species (which indicates the possibility of substantial chemistry effects); and (5) smaller amounts of volatile fission products are released from dispersed fuels than from alloy fuel. With regard to the fifth finding, at this stage the reverse was found[6] for the release of cesium from U_3Si_2–Al fuel. Significantly higher cesium release fractions were observed from silicide and aluminide dispersion fuels than from U–Al alloy fuels. The reason for this is not obvious and underscores the need to obtain experimental information with preirradiated fuel.

In a preliminary review of the ORNL and HEDL data, Lorenz[7] indicated that, for all the fission-product release experiments conducted with U–Al fuels, several shortcomings existed: (1) the chemical form and behavior of fission products released were not evaluated (especially for volatile iodine), (2) systematic release rate data were not obtained, (3) geometry effects were not evaluated, (4) the effect of the extent of oxidation on release was not studied, and (5) significant differences were observed between the HEDL and ORNL data for release of cesium and tellurium.

In addition to the preceding, significant errors may be introduced by straightforward extrapolation of information from experiments conducted with small coupon samples to "deep" debris beds (e.g., for large U–Al reactor core debris, such as for production reactors); and in all the U–Al fuel fission-product release experiments, the maximum fuel temperature was limited to 1300 K. This limit would require uncertain extrapolation for analyzing important melt progression phenomena involving ablation of structures, such as stainless steel, which melt at about 1700 K (neglecting eutectic formations, potential exothermic reactions, and compound formation effects when aluminum reacts with iron, nickel, and chromium, which also need to be taken into account).

MODELING OF FISSION-PRODUCT RELEASE FROM U–Al FUELS

The shortcomings of the experimental data obtained so far were noted while embarking on an effort to develop mathematical representations for modeling volatile fission-product releases from heated U–Al fuels. Recently, results of preliminary efforts at parametric modeling for fission-product release and bubbling from large pools have been reported.[8–10] Overall analysis of the data clearly showed that the development of mechanistic models for predicting releases for iodine, cesium, and tellurium species necessitates a good understanding of the fission-product chemistry under various conditions. This clearly represents a substantial undertaking. No mechanistic models for analyzing fission-product release from U–Al fuels have been developed. Therefore, until such models are completed and validated, empirical correlations should be crafted on the basis of analysis of characteristic trends, phenomenological considerations, and regression analysis for incorporation into such codes as MELCOR (Ref. 11).

Mathematical correlations–models were formulated with data from the experimental programs mentioned previously. The predicted releases of fission products from U–Al fuels account for the effects of release rate variation with ambient medium, burnup, time, and temperature. Modeling aspects and related assumptions for each of these effects are given in the text that follows.

Transient Fission-Product Release

As mentioned previously, none of the experimental programs investigated transient variations for fission-product release. Upon studying the Parker et al. data,[3] however, a substantial and unique dependence on time and temperature is indicated. Salient aspects of these data are shown in Fig. 2, where the ratio of releases over 3600 to 120 seconds are plotted at various temperatures. The significant variations with time and temperature for individual species are evident. As shown, much of the iodine release occurs over a relatively short duration and over a

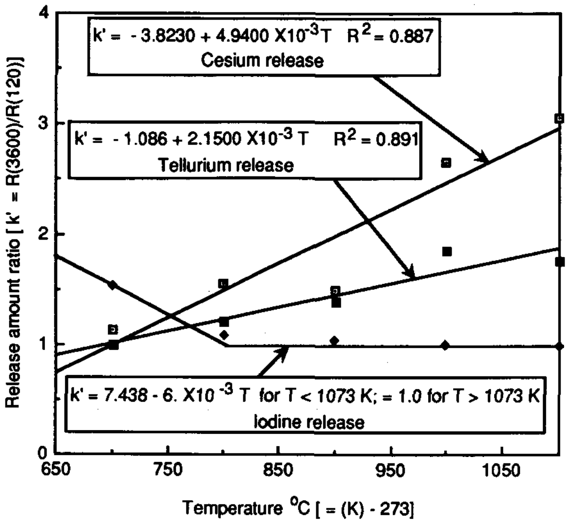


Fig. 2 Variation of ratio of release amount after 3600 seconds to that after 120 seconds vs. temperature.

wide range of temperatures, whereas this is not the case for cesium and tellurium. The data further indicate that assumptions of constant release rate (i.e., with time) at a given temperature are not realistic and may, in some circumstances, lead to nonconservative results (e.g., delayed release effects).

Another noteworthy aspect derived from Fig. 2 deals with potential inaccuracies resulting from modeling release fractions [F(t,T)] with time (t) and temperature (T) on the basis of the well-known CORSOR (i.e., first-order exponential variation) formulation written as

$$F(t,T) = 1 - e^{-kt} \quad (1)$$

The so-called rate constant k in Eq. 1 is obtained from data taken over a certain time frame. It should be recognized (in all fairness) that the CORSOR model is basically a correlation developed for oxidic power reactor fuels[12] and was never intended for general use. There was no reason to believe that it would work for U–Al fuels. For the confirmation of this belief, a test was conducted to evaluate the appropriateness of using the CORSOR form for capturing U–Al fuel transient fission-product release over 120 and 3600 seconds, as observed by Parker et al.[3] Results of this exercise are shown in Table 2. As noted in Table 2, if the rate constant is based on data taken over 120 seconds, Eq. 1 vastly overpredicts data taken over 3600 seconds and vice versa. The results of this exercise clearly indicate that the first-order exponential rate law given in Eq. 1 is inappropriate. A different approach is necessary for capturing the time dependence of volatile fission-product releases from heated U–Al fuels coupled with additional data for guidance and/or confirmation.

An alternate approach for capturing the time dependence was developed as shown in the ORNL data. The implicit assumption in this model is that the time dependence of release would be linear between 0 and 120 seconds and also between 120 and 3600 seconds and beyond. Thereafter, the general formulation for predicting percentage release for each volatile fission-product species (i.e., noble gases, iodine, cesium, and tellurium) is set up as

$$R(t,T) = R(120,T) \times t/120 \quad \text{(for t < 120 seconds)} \quad (2)$$

and

$$R(t,T) = R(120,T) + R(120,T) \times [k'(T) - 1.0] \times (t - 120)/3480 \quad \text{(for t >120 seconds)} \quad (3)$$

where R(120,T) is the amount (%) released on the basis of data taken over 120 seconds (including effects of ambient and burnup), T is the fuel temperature, and k'(T) is a temperature-dependent ratio of the release on the basis of data taken over 3600 seconds to that collected over 120 seconds. Specifically, k'(T) = 1 implies constant release amount over time, k'(T) < 1 implies that the release amount decreases with time, and k'(T) >1 implies that the release amount increases with time. Expressions for k'(T) variation with temperature for the various species are given in Fig. 2. Expressions for R(120,T) must be obtained from prototypical test results. In the absence of additional prototypical data and fission-product chemistry analysis, it is assumed that the time and temperature dependence of the release fractions for volatile fission-product species, as observed for U–Al alloy fuel in air, would be the same for other U–Al reactor fuels and different ambient conditions.

Variation of Fission-Product Release with Burnup

An important question for safety analysis deals with the variation of fission-product release with burnup. Analysis of ORNL data taken with samples of burnup up to 25 isotopic (^{235}U) percent seems to indicate a saturation effect after rising sharply from trace levels to about 3 isotopic (^{235}U) percent. This is especially true for the noble gases and iodine species. It is less clear for releases

Table 2 Performance of CORSOR Release Model Predictions for U–Al Reactor Fuels[a]

		Experiment				Calculations: CORSOR fractional release predictions			
		Collection time, 2 min		Collection time, 60 min		k = k1		k = k2	
Element	Temperature, °C	Fractional release	Rate constant (k1)	Fractional release	Rate constant (k2)	2 min	60 min	2 min	60 min
Cesium	900	0.06	3.20×10^{-2}	9.20×10^{-2}	1.61×10^{-3}	6.20×10^{-2}	8.53×10^{-1}	3.20×10^{-3}	9.20×10^{-2}
	1100	0.12	6.62×10^{-2}	3.78×10^{-1}	7.90×10^{-3}	1.24×10^{-1}	9.80×10^{-1}	1.56×10^{-2}	3.78×10^{-1}
Iodine	800	0.79	7.71×10^{-1}	8.50×10^{-1}	3.16×10^{-2}	7.86×10^{-1}	1.00×10^{0}	6.10×10^{-2}	8.50×10^{-1}
	1100	0.98	2.07×10^{0}	9.80×10^{-1}	6.52×10^{-2}	9.84×10^{-1}	1.00×10^{0}	1.22×10^{-1}	9.80×10^{-1}
Tellurium	800	0.00	1.00×10^{-3}	7.00×10^{-3}	1.17×10^{-4}	2.00×10^{-3}	5.80×10^{-2}	2.34×10^{-4}	7.00×10^{-3}
	1000	0.10	5.10×10^{-2}	1.66×10^{-1}	3.25×10^{-3}	9.70×10^{-2}	9.53×10^{-1}	6.50×10^{-3}	1.66×10^{-1}
	1100	0.45	2.97×10^{-1}	7.84×10^{-1}	2.55×10^{-2}	4.48×10^{-1}	1.00×10^{0}	4.90×10^{-2}	7.84×10^{-1}

[a]Units of rate constants k1 and k2 are 1/min each.

of cesium and tellurium, however, primarily because of data scatter that makes extrapolation beyond 24 isotopic (^{235}U) percent unreliable.

As noted from Table 1, the HEDL data taken for U–Al alloy specimens had a burnup of about 52 isotopic (^{235}U) percent. If we now superimpose the HEDL data with the ORNL data and assume the validity of combining the two data bases (i.e., assume absence of experimental biases), the effect of increasing burnup with temperature is shown in Fig. 3 to be highly nonlinear. The saturation effect is clear for iodine releases, with the same trend observed for the noble gases. For cesium and tellurium releases, however, a form of threshold effect is indicated whereby release amounts rise sharply, somewhere between 24 and 52 isotopic (^{235}U) percent burnup. A study of JAERI data taken with samples of 25 and 65 isotopic (^{235}U) percent burnup also displayed significant dependence on fuel burnup. This can be clearly seen from Fig. 1 (for noble gas release) and from Fig. 4 for volatile fission-product releases from U_xSi_y–Al fuel.

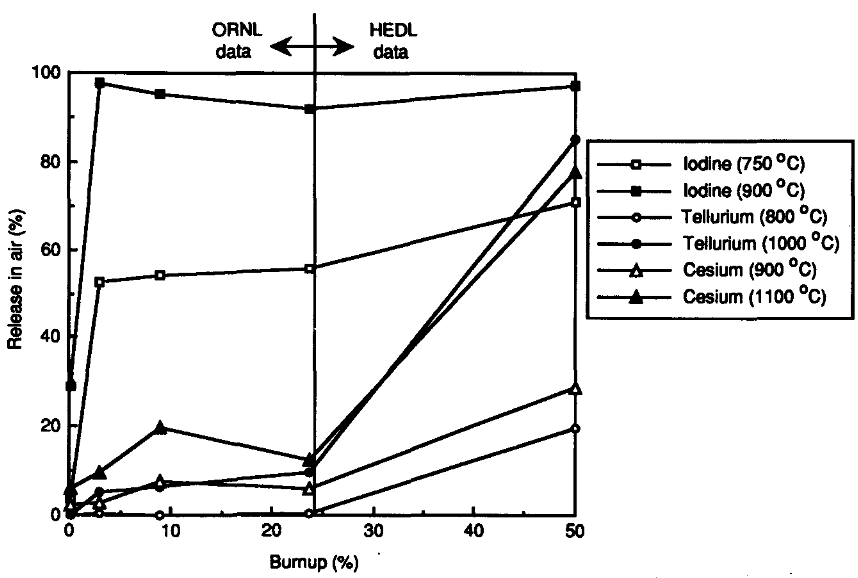


Fig. 3 Variation of volatile species release from U–Al alloy fuel with burnup.

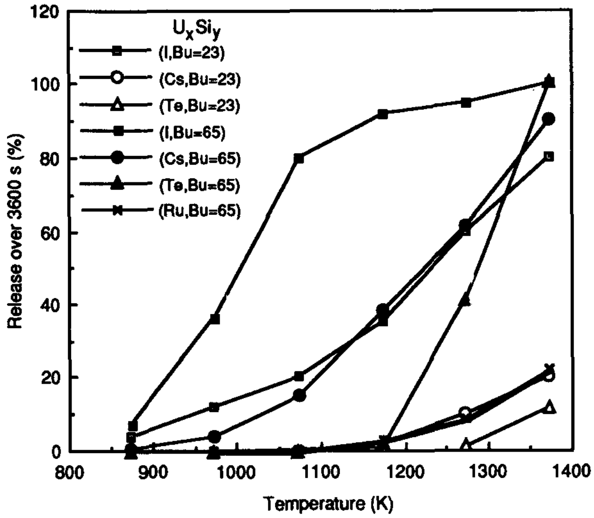


Fig. 4 Variation of fission-product release with temperature for U_xSi_y–Al fuel (in air).

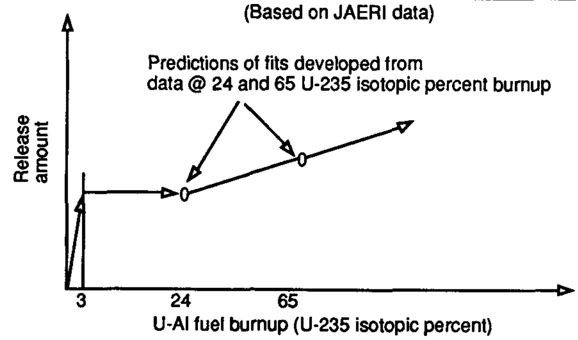
As concluded from discussions[13] with ORNL experts, there is no conclusive reason that can be cited as the cause for such a threshold effect. It is well known, however, that increasing burnup does increase fission-product pressure buildup in the fuel matrix. Such a pressure loading may then cause "burst" releases leading to selective threshold effects, as seen for the cesium and tellurium species, which usually display less volatility than the noble gases or iodine species. In the absence of a well-qualified mechanistic model backed with systematically obtained data for evaluating burnup dependence for fuels other than U–Al alloy fuel, it is recommended that for dispersion fuels (i.e., for U_3O_8–Al, U_3Si_2–Al, and U_xSi_y–Al) the effect of burnup be accounted for by linear interpolation as depicted in Table 3 [i.e., for obtaining estimates of release amounts between a burnup of 0 and 3 isotopic (^{235}U) percent, conduct a linear interpolation between predictions from fits to appropriate JAERI data at 24 isotopic (^{235}U) percent]. This prescription conservatively assumes (on the basis of Parker's data) that the first major jump in fission-product release occurs from trace levels to 3 isotopic (^{235}U) percent burnup and then levels off thereafter up to 24 isotopic (^{235}U) percent. For burnups between 24 and 65 isotopic (^{235}U) percent, inter-

Table 3 Burnup and Ambient Multipliers for Various Volatile Species[a]

Fission-product species	Burnup (Bu) multiplier equations (for U–Al dispersed and alloy fuels only)	Suggested multiplier for steam ambient: U–Al dispersed or alloy fuels	U_3O_8/U_3Si_2–Al fuels
Iodine	$0.25 \times (Bu + 1)$ for $Bu < 3\%$ 1.0 for $Bu > 3\%$	1.00	2.00
Cesium	$0.03 \times Bu + 0.25$ for $Bu < 25\%$ $0.17 \times (Bu - 25) + 1$ for $Bu > 25\%$	1.25	1.00
Tellurium	$0.32 \times Bu + 0.03$ for $Bu < 3\%$ 1.0 for $3\% < Bu < 25\%$ $0.4 \times (Bu - 25) + 1$ for $Bu > 25\%$	1.50	1.00
Noble gases	1.0 for all Bu	1.00	1.00

[a]Suggested interpolation path for U_3O_8-Al, U_3Si_2-Al, and U_xSi_y-Al dispersion fuels

(Based on JAERI data)



polate between predictions from fits to appropriate JAERI data at 24 and 65 isotopic (^{235}U) percent, respectively. For burnups greater than 65 isotopic (^{235}U) percent, conduct a linear extrapolation with a gradient term developed from predictions of fits from appropriate JAERI data at 24 and 65 isotopic (^{235}U) percent. Recall that, for U_3O_8–Al fuel, data were obtained from only one burnup level [i.e., 52 isotopic (^{235}U) percent]. For this fuel it is recommended that burnup multipliers be developed with the preceding prescription that uses a gradient term developed from the U_3Si_2–Al data fits.

Variations of Fission-Product Releases with Environment

For practical reactor accident analysis applications, fission-product release would need to be modeled under either dry (i.e., predominantly air) or wet (i.e., steam–air) ambient media. The theoretical case of release in inert media may be useful for benchmarking mechanistic models and for demonstrating the effect of oxidizing environments. For the development of "empirical" correlations or models, we ignore the effect of inert ambient media. However, the effect of ambient medium on the amount of release is dramatized by including selected data (taken by Parker et al. for U–Al alloy fuel in inert media), as shown in Fig. 5.

Figure 5 indicates that the presence of steam can indeed have a significant effect on the release of cesium and tellurium species. The impact of steam versus air environment on noble gas and iodine releases was found to be insignificant and well within experimental uncertainties. A similar conclusion can be drawn from analysis of the HEDL data base despite the large amounts of data scatter. This is true except for iodine release from U_3O_8–Al fuel samples where a significant enhancement of release was observed when releases occurred in a steam environment.

The actual chemical environment experienced by the heated U–Al fuel may be different from that of the bulk gas composition. This is true when oxidizing atmospheres cause significant chemical reactions with the cladding material. Specifically, in the case of a steam environment, hydrogen is produced because of the aluminum–steam reaction. If the resulting hydrogen is entrapped on the fuel surface region, it can lead to the formation of a reducing environment for the various fission products. As is known,[14] Zircaloy–steam interactions produce significant quantities of hydrogen even before Zircaloy becomes molten. Fortunately, aluminum–steam reactions[15] are far less energetic, even if the aluminum is molten. Aluminum does not begin to ignite unless the molten aluminum temperature reaches values close to 1650 K (under atmospheric conditions). As a word of caution, the analyst should note that the aspect of aluminum–steam ignition and resulting hydrogen production is strongly dependent on certain key parameters (i.e., aluminum particle size, steam pressure, and aluminum temperature). This aspect should be taken into account before drawing conclusions on the effect of environment on fission-product chemistry and release.

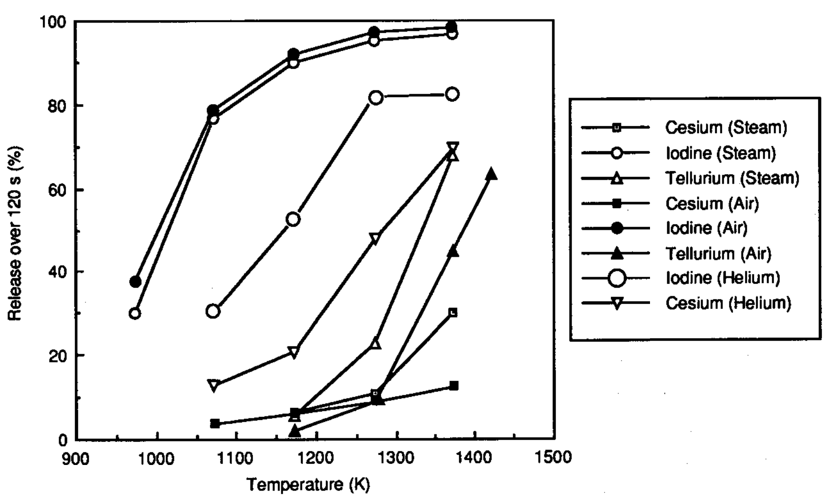


Fig. 5 Variation of volatile fission-product releases from U–Al alloy fuel in steam, air, and helium with temperature (ORNL data).

Because of the significant variations of release amounts in air versus steam, correlation constants were developed for air, and separately for steam ambients, when data were available. As mentioned previously, however, the JAERI data were obtained in air alone. For such instances, it is advised that correlations developed for air media be used for calculating releases in steam, along with suitable multipliers developed from the ORNL and HEDL data bases. One reason for this prescription is related to the fact that Parker et al.[3] used a silica crucible to melt fuel specimens. Hence chemistry aspects related to the presence of a silicon component were accounted for. An overall analysis of the ORNL and HEDL data bases suggested the use of multipliers for predicting fission-product releases in steam for the various fuel types. Suggested multipliers are shown in Table 3.

Development of Correlation Library from Experimental Data

In this section, mathematical formulations are presented for the parameters R(120,T), R(3600,T), and k'(T), as described earlier in Eq. 2. These are polynomial or exponentially based expressions that are different for each individual species and fuel type investigated. The expressions were developed via regression and trend analysis in three basic categories:

Category	Correlation form for R(120,T) or R(3600,T)	
I	$A \times \exp(BT)$ (i.e., exponential form)	(4)
II	$A \times \exp(-Q/RT)$ (i.e., Arrhenius form)	(5)
III	Trend line analysis (e.g., polynomial fit)	(6)

where A = dimensionless constant
B = constant based on one or multiple temperature ranges, K^{-1}
Q = activation energy, kcal / mol
R = gas constant (= 0.00199 kcal / mol / K)
T = fuel temperature, K

Note that such codes as MELCOR refer to the forms of Eqs. 4 and 5 as CORSOR and CORSOR-M formulations, respectively. Therein, A is a rate constant with dimensions of min^{-1} and does not vary with time; that is, A (in MELCOR) = A in Eq. 4 or 5 divided by the fuel heating time (minutes). This important distinction should be kept in mind when applying correlations presented in this section directly to such codes as MELCOR. Note that, as demonstrated in Table 2, a rate constant developed in this manner may give rise to large amounts of over- or under-predictions. Hence caution is advised and a recommendation is made that code developers allow for modification of this predictive scheme. Note also that, in the original CORSOR (i.e., Eq. 4) formulation, the Celsius scale is used for specifying the fuel temperature. To conform with SI units (for this study), however, Eq. 4 coefficients are developed in conjunction with the fuel temperature specified in Kelvin. It was further found that the built-in (i.e., default) values for MELCOR's CORSOR and CORSOR-M models for power reactor fuels predicted no release for U–Al reactor fuels for temperatures up to 1373 K. Hence new coefficients were developed for various temperature ranges, individual species, and ambient media as appropriate for direct use in severe accident analysis codes.

Curve fitting was performed systematically in several stages. First, Category I (i.e., exponential) and piecewise linear fits (part of Category III) were derived. Thereafter, Categories II and III fits were derived. Where necessary, upon trend analysis, coefficients were obtained over two temperature ranges also. Such a breakup was found necessary for the iodine species in the ORNL and HEDL data bases and for iodine and cesium species in the JAERI data base. For the iodine species, a piecewise linear formulation was found to properly represent observed trends, whereas an exponential representation captured trends well enough for most of the other species. A linear profile best captured cesium release data from U_3Si_2–Al fuel. The various correlation constants in different categories are shown in Tables 4 to 6. Alongside the constants is also given the so-called "coefficient of correlation (R^2)," which indicates the goodness of fit. The improvement in R^2 with breakup of the temperature ranges is evident. Note that the large degree of data scatter in the HEDL data, especially for the U_3O_8–Al specimens, contributes to the general lowering of R^2.

Finally, it should be realized that the excellent correlation of fits with data for some of the species (e.g., tellurium in the HEDL data base) is primarily due to the extremely small number of data points (two in this case). Therefore caution is advised in this instance. More data are needed. The ORNL data, however, do indicate exponential variation with temperature for tellurium release.

Table 4 Exponential Form Correlation Coefficients Summary

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range variation: Number	Range variation: Temperature range, K	A	B	Coefficient of correlation
ORNL									
(Parker et al.)[3]	UAl_4 alloy	24.00	Cs	Steam	1	>873	7.63×10^{-4}	7.65×10^{-3}	0.97
				Air	1	>873	6.05×10^{-2}	3.89×10^{-3}	0.99
			I	Steam–air		>1050	1.17×10^{-6}	1.70×10^{-2}	0.87
					2	>1050	3.73×10^{1}	7.30×10^{-4}	0.83
			Te	Steam	1	>873	2.94×10^{-6}	1.24×10^{-2}	1.00
				Air	1	>873	1.16×10^{-7}	1.43×10^{-2}	0.99
HEDL									
(Woodley et al.)[4]	UAl_4 alloy	52.00	Cs	Steam	1	>873	1.27×10^{0}	3.20×10^{-3}	0.63
				Air	1	>873	3.56×10^{-1}	4.08×10^{-3}	0.72
			Te	Steam	1	>873	4.86×10^{-10}	2.30×10^{-2}	1.00
				Air	1	>873	2.59×10^{-6}	1.40×10^{-2}	0.96
			I	Steam	1	>873	5.18×10^{-1}	4.24×10^{-3}	0.83
					2	<1100	3.14×10^{-4}	1.12×10^{-2}	0.91
						>1100	3.83×10^{1}	7.21×10^{-4}	0.80
				Air	1	>873	2.45×10^{-1}	4.79×10^{-3}	0.50
					2	<950	1.00×10^{-13}	3.52×10^{-2}	1.00
						>950	4.33×10^{1}	5.94×10^{-4}	0.49
	U_3O_8–Al		Cs	Steam	1	>873	4.81×10^{-1}	3.45×10^{-3}	0.19
				Air	1	>873	1.21×10^{-4}	1.01×10^{-2}	0.60
			Te	Steam	1	>873	5.12×10^{-5}	1.06×10^{-2}	1.00
				Air	1	>873	1.69×10^{-4}	1.02×10^{-2}	0.72
			I	Steam	1	>873	1.26×10^{-2}	7.92×10^{-3}	0.32
					2	<950	7.71×10^{-17}	4.31×10^{-2}	0.65
						>950	4.27×10^{1}	5.64×10^{-4}	0.12
				Air	1	>873	1.08×10^{0}	3.34×10^{-3}	0.65
					2	<1050	3.94×10^{-10}	2.42×10^{-2}	0.97
						>1050	1.97×10^{0}	2.88×10^{-3}	0.75
JAERI									
(Saito et al.)[5]	U_3Si_2–Al	23.00	Cs	Air	1	>873	2.28×10^{-2}	5.94×10^{-3}	0.92
			I	Air	2	<1100	7.09×10^{-6}	1.47×10^{-2}	1.00
						>1100	1.97×10^{1}	1.09×10^{-3}	0.98
			Te	Air	1	>873	4.04×10^{-8}	1.51×10^{-2}	1.00
			Noble gases	Air	1	>850	4.27×10^{-3}	8.17×10^{-3}	0.46
					2	850 to 873	1.69×10^{-42}	1.14×10^{-1}	1.00
						>873	3.30×10^{1}	8.53×10^{-4}	0.81
	UAl–dispersed	23.00	Cs	Air	1	>873	1.02×10^{-6}	1.33×10^{-2}	0.65
					2	<1175	2.08×10^{-20}	4.17×10^{-2}	1.00
						>1175	7.46×10^{0}	1.24×10^{-3}	0.99
			I	Air	1	>873	1.31×10^{0}	2.79×10^{-3}	0.98
			Te	Air	1	>873	7.70×10^{-2}	2.88×10^{-3}	1.00
	U_xSi_y–Al	23.00	Cs	Air	1	>873	1.47×10^{-3}	6.93×10^{-3}	1.00
			I	Air	1	>873	3.39×10^{-2}	5.83×10^{-3}	0.96
					2	<1150	3.92×10^{-3}	8.06×10^{-3}	0.96
					2	>1150	2.86×10^{-1}	4.15×10^{-3}	0.97

Table 4 (Continued)

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range variation		A	B	Coefficient of correlation
					Number	Temperature range, K			
			Te	Air	1	>853	4.78×10^{-12}	2.08×10^{-2}	1.00
			Noble gases	Air	1	>850	7.44×10^{-3}	7.74×10^{-3}	0.39
					2	850 to 873	3.76×10^{-46}	1.24×10^{-1}	1.00
						>873	3.79×10^{1}	7.46×10^{-4}	1.00
JAERI (Saito et al.)[5]	U_3Si_2–Al	65.00	Cs	Air	1	>873	1.42×10^{-2}	6.79×10^{-3}	0.89
					2	<1073	6.96×10^{-5}	1.23×10^{-2}	1.00
						>1073	3.03×10^{0}	2.55×10^{-3}	1.00
			I	Air	2	850 to 1050	4.64×10^{-3}	9.27×10^{-3}	0.92
						>1050	4.48×10^{1}	5.91×10^{-4}	0.94
					1	>850	1.24×10^{0}	3.46×10^{-3}	0.67
			Te	Air	1	>1050	4.81×10^{-11}	2.10×10^{-2}	0.87
			Ru	Air	1	>1100	6.61×10^{-7}	1.30×10^{-2}	0.88
					2	<1173	9.03×10^{-14}	2.72×10^{-2}	1.00
						>1173	1.72×10^{-3}	6.98×10^{-3}	1.00
			Noble gases	Air	1	>850	1.28×10^{-2}	7.33×10^{-3}	0.34
					2	850 to 873	4.75×10^{-49}	1.32×10^{-1}	1.00
						>873	7.23×10^{1}	2.51×10^{-4}	0.75
	UAl–dispersed	65.00	Cs	Air	1	>873	4.65×10^{-2}	5.92×10^{-3}	0.89
			I	Air	1	>873	1.10×10^{-5}	1.45×10^{-2}	0.96
					2	873 to 975	8.38×10^{-8}	1.99×10^{-2}	1.00
						>975	3.31×10^{1}	8.13×10^{-4}	0.96
			Te	Air	1	>873	1.88×10^{-18}	3.27×10^{-2}	1.00
			Ru	Air	1	>873	1.06×10^{-7}	1.23×10^{-2}	0.92
	U_xSi_y–Al	65.00	Cs	Air	1	>873	3.20×10^{-4}	9.54×10^{-3}	0.93
					2	873 to 1073	1.16×10^{-6}	1.53×10^{-2}	0.99
						>1073	2.45×10^{-1}	4.31×10^{-3}	1.00
			I	Air	1	>850	2.69×10^{-1}	4.67×10^{-3}	0.70
					2	850 to 1050	1.94×10^{-4}	1.22×10^{-2}	0.96
						>1150	3.88×10^{-1}	7.02×10^{-4}	0.90
			Te	Air	1	>873	1.80×10^{-8}	1.62×10^{-2}	0.93
					2	873 to 1173	7.40×10^{-6}	1.00×10^{-2}	0.96
						>1173	4.83×10^{-4}	8.92×10^{-3}	1.00
			Ru	Air	1	>873	3.64×10^{-8}	1.50×10^{-2}	0.96
						873 to 1100	1.07×10^{-5}	9.16×10^{-3}	1.00
						>1100	3.19×10^{-5}	9.80×10^{-3}	1.00
			Noble gases	Air	1	>850	1.06×10^{-2}	7.46×10^{-3}	0.36
					2	850 to 873	5.01×10^{-48}	1.30×10^{-1}	1.00
						>873	5.58×10^{1}	4.45×10^{-4}	0.85
ORNL (Shibata et al.)[2]	UAl_x–dispersed	62.00	Noble gases	Helium	1	>850	9.89×10^{-5}	1.52×10^{-2}	0.62

Table 5 Arrhenius Correlation Coefficients

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range number	Variation range	In(A)	Q, kcal/mol	Coefficient of correlation
ORNL									
(Parker et al.)[3]	UAl_4 alloy	24.00	Cs	Steam	1	>873	12.17	24.26	0.95
				Air	1	>873	6.71	11.45	1.00
			I	Steam–air	2	933 to 990	69.25	128.55	1.00
						>990	6.94	5.98	0.75
			Te	Steam	1	>873	18.79	39.76	0.99
				Air	1	>873	20.94	47.22	1.00
HEDL									
(Woodley et al.)[4]	UAl_4 alloy	52.00	Cs	Steam	1	>873	7.48	8.00	0.68
				Air	1	>873	8.42	10.75	0.75
			Te	Steam	1	>873	25.90	48.10	1.00
				Air	1	>873	17.37	32.06	0.98
			I	Steam	1	>873	8.92	10.66	0.89
					2	873 to 970	45.43	80.73	1.00
						>970	5.41	2.13	0.83
				Air	1	>873	9.64	12.39	0.59
					2	873 to 950	35.81	60.77	1.00
						>950	5.20	1.69	0.59
	U_3O_8–Al		Cs	Steam	1	>873	7.15	8.91	0.19
				Air	1	>873	8.26	11.44	0.48
			Te	Steam	1	>873	13.09	24.32	1.00
				Air	1	>873	14.56	26.05	0.80
			I	Steam	1	>873	13.24	19.09	0.40
					2	873 to 975	42.68	73.33	0.65
						>975	5.10	1.57	0.17
				Air	1	>873	7.92	9.06	0.64
					2	873 to 1000	25.17	44.74	0.97
						>1000	7.92	9.06	0.64
JAERI									
(Saito et al.)[5]	U_3Si_2–Al	23.00	Cs	Air	1	>873	9.55	14.55	0.97
					2	873 to 1100	11.10	17.51	0.97
						>1100	6.63	7.21	0.99
			I	Air	1	>873	11.25	17.14	0.90
					2	873 to 1100	16.62	27.30	1.00
						>1100	5.76	3.51	0.99
			Te	Air	1	>873	19.73	44.10	0.99
			Noble gases	Air	2	850 to 873	97.29	163.10	1.00
						873 to 1373	6.73	5.27	0.79
	UAl-dispersed		Cs	Air	1	>873	19.47	40.91	0.71
					2	873 to 1175	48.09	104.08	1.00
						>1175	5.16	3.96	1.00
			I	Air	1	>873	10.03	15.33	0.80
					2	873 to 950	25.61	43.78	1.00
						>950	6.78	7.42	0.98

Table 5 (Continued)

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range number	Variation range	In(A)	Q, kcal/mol	Coefficient of correlation
			Te	Air	1	>873	5.05	10.01	1.00
	U_xSi_y–Al		Cs	Air	1	>873	20.08	45.81	0.98
			I	Air	1	>873	9.61	14.10	0.99
			Te	Air	1	>873	28.96	72.32	1.00
			Noble gases	Air	2	850 to 873	106.20	177.71	1.00
						873 to 1373	5.89	3.23	0.88
JAERI (Saito et al.)[5]	U_3Si_2–Al	65.00	Cs	Air	1	>873	11.07	16.81	0.96
			I	Air	1	>850	8.19	8.91	0.78
					2	850 to 973	16.05	23.43	1.00
						>973	5.21	1.75	0.97
			Te	Air	1	>1100	30.06	68.34	0.90
			Ru	Air	1	>1075	17.78	38.91	0.92
					2	1075 to 1173	30.97	68.03	1.00
						>1173	11.34	22.27	1.00
			Noble gases	Air	2	850 to 873	113.28	189.30	1.00
						873 to 1373	5.13	1.31	0.83
	UAl-dispersed	65.00	Cs	Air	1	>873	10.30	14.67	0.96
			I	Air	1	>873	11.06	16.21	0.87
					2	873 to 1073	16.96	27.36	0.98
						>1073	5.57	2.62	0.97
			Te	Air	1	>873	30.05	72.08	0.91
			Ru	Air	1	>873	10.78	28.43	0.86
	U_xSi_y–Al	65.00	Cs	Air	1	>873	13.39	23.41	0.98
			I	Air	1	>850	9.43	11.99	0.81
					2	873 to 1073	15.21	22.89	0.98
						>1073	5.09	1.33	0.97
			Te	Air	1	>873	17.44	37.30	0.87
					2	873 to 1173	8.38	19.98	0.92
						>1173	15.96	31.01	1.00
			Ru	Air	1	>873	17.81	39.93	0.97
					2	873 to 1073	7.31	19.04	1.00
						>1073	14.55	31.36	0.99
			Noble gases	Air	2	850 to 873	110.79	185.21	1.00
						873 to 1373	5.39	1.97	0.88
ORNL (Shibata et al.)[2]	UAl_x-dispersed		Noble gases	Helium	1	>850	17.85	23.97	0.65

Table 6 Polynomial Form Correlations Summary

Institution (researchers)	Release time, s	Burnup, %	Fuel	Species	Ambient	Range variation Number	Range	Release formula = f[Temperature (K)] %
ORNL (Parker et al.)[3]	120.00	24.00	UAl_4 alloy	I	Steam–air	2	875 to 1125 >1125	0.34 – 297.5 0.075T + 0.62
HEDL (Woodley et al.)[4]	120.00	53.00	UAl_4 alloy	I	Steam	2	875 to 1125 >1125	0.34T – 297.5 0.075T + 0.62
					Air	2	875 to 975 >975	0.8T – 700 0.05T + 31.25
			U_3O_8–Al	I	Steam	2	875 to 1125 >1125	0.34T – 297.5 0.075T + 10
					Air	1	>875	0.222T – 194.4
JAERI (Saito et al.)[5]	3600.00	23.00	U_3Si_2–Al	Cs	Air	1	>883	–96.29 + 0.109T
				Noble gases	Air	1	825 to 1373	$-2138.8 + 5.31T - 4.21 \times 10^{-3}T^2 + 1.115 \times 10^{-6}T^3$
			UAl–dispersed	Cs	Air	2	1075 to 1175 >1175	0.3T – 322 0.0495T – 26
				I		1	>873	–90.05 + 0.1085T
			U_xSi_y–Al	Cs		1	>1175	0.105T – 123
				I		2	825 to 1100 >1100	0.0909T – 75 0.214T – 211
				Te	Air	1	>1250	0.107T – 133
				Noble gases	Air	2	823 to 873 873 to 1373	T – 823 $-839.2 + 2.01T - 1.438 \times 10^{3}T^2 + 3.426 \times 10^{-7}T^3$
		65.00	U_3Si_2–Al	Cs	Air	2	840 to 973 >973	0.09T – 75.57 0.22T – 202.06
				I	Air	2	850 to 1050 >1050	–291.22 + 0.35T 26.46 + 0.054T
				Ru	Air	1	>1086	–86.412 + 0.0796T
				Noble gases	Air	2	823 to 875 875 to 1373	–1234.5 + 1.5T $-666.25 + 1.7905T - 1.393 \times 10^{-3}T^2 + 3.611 \times 10^{-7}T^3$
			UAl-dispersed	Cs	Air	2	827 to 1000 >1000	0.11T – 91 0.264T – 243.3
				I	Air	2	857 to 1100 >1100	–162.87 + 0.19T –2.1417 + 0.075T
					Air	1	>873	–226.31 + 0.26T
				Ru	Air	1	1153 to 1373	0.01T – 11.53
			U_xSi_y–Al	Cs	Air	3	870 to 973 973 to 1073 >1073	0.33T – 287.4 –103 + 0.11T –253 + 0.25T
				I	Air	2	850 to 1100 >1100	–311.6 + 0.365T 14.701 + 0.063T
				Te	Air	2	1030 to 1178 >1178	0.0077T – 7.932 –671.4 + 0.57T
				Noble gases	Air	2	823 to 873 873 to 1373	1.3T – 1069.9 $-830.05 + 2.165T - 1.694 \times 10^{-3}T^2 + 4.44 \times 10^{-7}T^3$
ORNL (Shibata et al.)[2]	1800.00	62.00	UAl_x-dispersed	Noble gases	Helium	1	823 to 923	$R(120,T) = -10216 + 22.57T - 1.23 \times 10^{-2}T^2$

Table 7 Suggested Correlation Forms Summary

Institution (researchers)	Burnup, %	Fuel	Species	Ambient	Number of temperature ranges	Suggested formulations
ORNL (Parker et al.)[3]	24	UAl_4 alloy	Cs	Steam	1	Exponential (see Table 4)
				Air	1	Exponential (see Table 4)
			I	Steam	2	Polynomial (see Table 6)
				Air	2	Polynomial (see Table 6)
			Te	Steam	1	Exponential (see Table 4)
				Air	1	Exponential (see Table 4)
HEDL (Woodley et al.)[4]	52	UAl_4 alloy	Cs	Steam	1	Arrhenius (see Table 5)
				Air	1	Arrhenius (see Table 5)
			Te	Steam	1	Arrhenius (see Table 5)
				Air	1	Arrhenius (see Table 5)
			I	Steam	2	Polynomial (see Table 6)
				Air	2	Polynomial (see Table 6)
	52	U_3O_8–Al	Cs	Steam	1	Arrhenius (see Table 5)
				Air	1	Arrhenius (see Table 5)
			Te	Steam	1	Arrhenius (see Table 5)
				Air	1	Arrhenius (see Table 5)
			I	Steam	2	Arrhenius (see Table 5)
				Air	2	Polynomial (see Table 6)
				Air	2	Polynomial (see Table 6)
JAERI (Saito et al.)[5]	23	U_3Si_2–Al	Cs	Air	1	Polynomial (see Table 6)
			I	Air	2	Polynomial (see Table 6)
			Te	Air	1	Polynomial (see Table 6)
			Noble gases	Air	1	Polynomial (see Table 6)
	23	UAl-dispersed	Cs	Air	2	Arrhenius (see Table 5)
			I	Air	1	Polynomial (see Table 6)
			Te	Air	1	Arrhenius (see Table 5)
			Noble gases	Air	1	Polynomial (see Table 6)
	23	U_xSi_y–Al	Cs	Air	1	Polynomial (see Table 6)
			I	Air	2	Polynomial (see Table 6)
			Te	Air	1	Arrhenius (see Table 5)
			Noble gases	Air	2	Polynomial (see Table 6)
	65	U_3Si_2–Al	Cs	Air	1	Polynomial (see Table 6)
			I	Air	2	Polynomial (see Table 6)
			Te	Air	1	Arrhenius (see Table 5)
			Re	Air	2	Exponential (see Table 4)
			Noble gases	Air	2	Polynomial (see Table 6)
	65	UAl-dispersed	Cs	Air	1	Polynomial (see Table 6)
			I	Air	2	Arrhenius (see Table 5)

(Table continues on the next page.)

Table 7 (Continued)

Institution (researchers)	Burnup, %	Fuel	Species	Ambient	Number of temperature ranges	Suggested formulations
			Te	Air	1	Arrhenius (see Table 5)
			Ru	Air	2	Polynomial (see Table 6)
			Noble gases	Air	1	Polynomial (see Table 6)
	65	U_xSi_y–Al	Cs	Air	1	Polynomial (see Table 6)
			I	Air	2	Arrhenius (see Table 5)
			Te	Air	2	Arrhenius (see Table 5)
			Ru	Air	2	Arrhenius (see Table 5)
			Noble gases	Air	2	Polynomial (see Table 6)
ORNL (Shibata et al.)[2]	62	UAl_x-dispersed	Noble gases	Air	1	Polynomial (see Table 6)

Therefore exponential formulations for release of tellurium species from fuel specimens in the HEDL experiments are justified for use in the absence of more data.

In the final stage, a further analysis was conducted to evaluate a list of correlation formulations that best represented the data with a view toward the possible need for extrapolation. These are herein referred to as "suggested formulations" and are tabulated as such in Table 7.

When applying these correlations for evaluating iodine, cesium, tellurium, and ruthenium species release amounts, unless otherwise stated, the minimum temperature to be used is the aluminum-clad melting temperature, below which the release amounts should be set to zero. For noble gas releases, the corresponding minimum temperature should be set to coincide with the onset of blistering, which is approximately 833 K for U–Al alloy fuels.

CORRELATION STATISTICS

A statistical analysis was conducted to see how well the various categories of fits mentioned in the previous section captured the data bases on an individual and collective basis. Selected results of this analysis are shown in Table 8 and Fig. 6, which show that a large portion of the data is captured within an error band of about ±30% over the entire temperature range investigated. This is good, considering that inherent in the experimental data is a measurement uncertainty of about ±10 to 20%. The mean value of predicted-to-measured ratios is close to 1.0 for all categories, whereas the overall standard deviations are 0.22, 0.27, and 0.33 for the suggested forms, Categories II and I fits, respectively.

Table 8 also gives a breakdown of the mean and standard deviations for individual data bases. As noted therein, a large contributor to the increase of the overall standard deviation is attributable to data scatter in the HEDL observations. If the HEDL data are excluded in the statistical computations, the overall statistics improve significantly. As noted in Table 8, with the suggested correlation formulations, the new values for the mean and standard deviation are 1.0 and 0.09, respectively.

CONCLUSIONS

This article has synthesized, analyzed, and modeled the available fission-product release data for common U–Al reactor fuels and developed an extensive library of correlations for predicting release rates that may vary with time, as well as ambient environment, burnup, and temperature, subject to assumptions mentioned in the article. As has been described, despite the fine efforts made by past experimenters, several shortcomings exist in the data bases. These shortcomings mainly relate to lack of information on fission-product chemistry, geometry effects, and time dependence of rate of release. Further, in all the data bases the maximum fuel temperature was limited to 1300 K, and for a few cases the data base was extremely sparse. Several details (including a compilation of all data

Table 8 Correlation Statistics

Data/fuel	Burnup, %	Species	Exponential form Mean	Exponential form Standard deviation	Arrhenius form Mean	Arrhenius form Standard deviation	Suggested form Mean	Suggested form Standard deviation
HEDL data								
UAl/U_3O_8	52	Cs	1.09	0.67	1.09	0.47	1.09	0.47
UAl	52		1.04	0.33	1.04	0.31	1.04	0.31
U_3O_8	52		1.15	0.95	1.14	0.61	1.14	0.61
UAl/U_3O_8	52	I	0.96	0.24	0.96	0.23	0.99	0.24
UAl/U_3O_8	52	Te	1.11	0.48	1.07	0.37	1.07	0.37
ORNL data								
UAl	24 and 62	Cs, I, Te, and noble gases	0.97	0.20	0.95	0.20	1.00	0.09
JAERI data								
Dispersed UAl, U_3Si_2–Al, and	24	Cs, I, Te, and noble gases	1.04	0.26	1.04	0.26	1.03	0.10
U_xSi_y–Al	65	Cs, I, Te, and noble gases	0.98	0.19	0.98	0.18	1.00	0.08
All data	24 to 65	Cs, I, Te, and noble gases	1.01	0.33	1.01	0.27	1.02	0.22
ORNL/JAERI data	24 and 65	Cs, I, Te, and noble gases	0.99	0.20	0.99	0.20	1.00	0.09

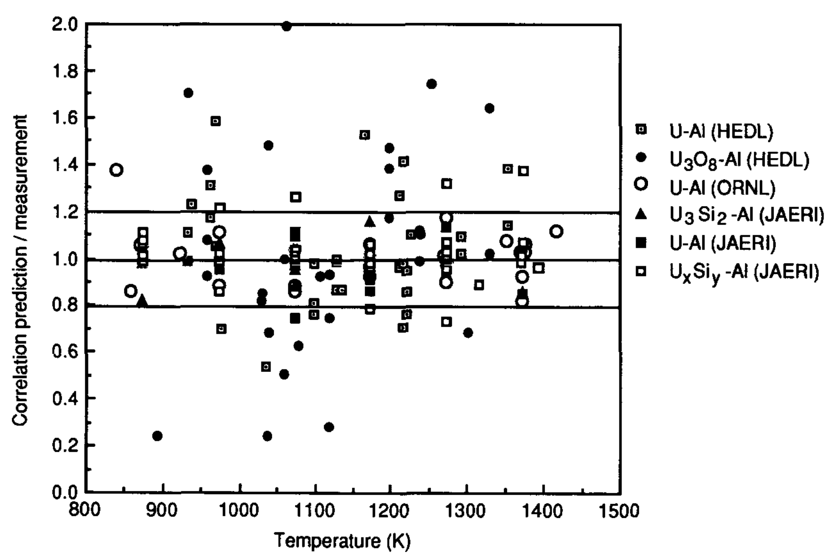


Fig. 6 Variation of suggested correlation predictions to measurements with temperature. The mean is 1.00 (σ, 0.09) excluding the HEDL data and 1.02 (σ, 0.22) if the HEDL data are included in the calculation.

bases cited in this article) that have been omitted can be found in the yet to be published report.[16] Such a library should be a valuable and useful reservoir of information to researchers and modelers in severe accident safety analysis of U–Al fueled reactors.

Finally, a few words of caution are in order. The library of correlations presented in this article is empirically based and as such can be used for severe accident analysis with confidence primarily when fuel conditions are within the range of experimental test parameters (from which correlation constants were developed). Extrapolation beyond specific test conditions can be quite risky.

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Control and Instrumentation

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Applications of a Surveillance and Diagnostics Methodology Using Neutron Noise From a Pressurized-Water Reactor[a]

By R. T. Wood,[b] L. F. Miller,[c] and R. B. Perez[c]

Abstract: *Two applications of a noise diagnostic methodology were performed with ex-core neutron detector data from a pressurized-water reactor (PWR). A feedback dynamics model of the neutron power spectral density was derived from a low-order whole-plant physical model made stochastic with the Langevin technique. From a functional fit to plant data, the response of the dynamic system to changes in important physical parameters was evaluated by a direct sensitivity analysis. In addition, changes in monitored spectra were related to changes in physical parameters, and detection thresholds using common surveillance discriminants were determined. A resonance model was developed from perturbation theory to give the ex-core neutron detector response for small in-core mechanical motions in terms of a pole-strength factor, a resonance asymmetry (or skewness) factor, a vibration damping factor, and a frequency of vibration. The mechanical motion parameters for several resonances were determined by a functional fit of the model to plant data taken at various times during a fuel cycle and were tracked to determine trends that indicated vibrational changes of reactor internals. In addition, the resonance model gave the ability to separate the resonant components of the power spectral density after the parameters had been identified. As a result, the behavior of several vibration peaks was monitored over a fuel cycle. The noise diagnostic methodology illustrated by these applications can be used in monitoring the condition of the reactor system. Early detection of degraded mechanical components or undesirable operating conditions by using such surveillance and diagnostic techniques would enhance plant safety.*

The investigation of stochastic fluctuations about the average (or d-c value) in detector signals from a nuclear power plant (reactor noise analysis) provides the opportunity to gain dynamic information about the reactor system without requiring the disturbance of the system by outside actions.[1,2] This capability arises because the fluctuations exhibited by state variables of the reactor system contain information about their origin and about the dynamic transmission properties of the reactor. Such fluctuations can be represented by noise descriptors that characterize the state of the power plant. Noise descriptors, such as power spectral densities (PSDs) and cross power spectral densities (CPSDs), display features (e.g., peaks and valleys) that are related to specific causative mechanisms, such as fuel vibrations, core barrel motion, thermal–hydraulic processes, and reactivity feedback effects.[3] These features define the plant signature. The low-frequency (0.001 to 1.0 Hz) behavior of pressurized-water-reactor (PWR) neutron noise is greatly affected by thermal–hydraulic feedback effects and the interrelated energy transport processes of the system. In the high-frequency range (1 to 20 Hz) and above, PWR neutron noise is dominated by vibration peaks resulting from the motion of reactor internals. Plant surveillance is accomplished by monitoring noise descriptors for changes in

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the plant signature, which indicate changes in the dynamic state of the plant.

Reactor monitoring and system condition diagnosis based on neutron and process variable noise have wide application in the international nuclear community. The Specialists Meeting on Reactor Noise (SMORN) symposia[4–9] present key developments in noise analysis and provide a clear indication of the state of the art. The nondisruptive nature of noise observations allows for frequent surveillance of the reactor's dynamic condition without interfering with normal plant operations. In fact, noise analysis permits automated, continuous, on-line surveillance based on pattern recognition methods to detect anomalous behavior.[1] In such surveillance systems, pattern recognition techniques are used to continuously monitor noise descriptors, obtained from various plant signals, for deviations from the plant's normal or baseline signature. Once a suspect descriptor is identified, the surveillance system records it to allow a later evaluation of the reactor's dynamic condition by a noise analyst. Consequently the altered state of the plant, which induced the change in the signature, may be diagnosed. However, diagnosis of power reactor noise is not an exact science because of the complexity of the feedback mechanisms and mechanical perturbations and because of the limited knowledge about the stochastic noise sources driving the behavior of reactor systems.[1,3] As a result, power reactor noise diagnostics tend to be qualitative in nature. Reactor noise diagnostics depend on observations resulting from correlation analyses of measured signals, prior knowledge of system behavior derived from experimental simulations and theoretical modeling, and intuition of the noise analyst.[1,10,11] Thus most diagnostic capabilities are not in a form that can be easily formalized or automated. For the successful development of an automated diagnostic system, a procedure must be devised to allow observed changes in a plant's signature to be characterized by systematically identifiable quantities.[1]

Research at the Oak Ridge National Laboratory (ORNL) has been directed toward addressing the need for tools that can be used for evaluating the diagnostic content of neutron PSDs from ex-core detectors at a PWR in a systematic way. The techniques involved include stochastic modeling of the dynamic processes, parameter estimation with plant noise data, sensitivity analyses for the detection of physically significant parameters with the adjusted models, and trending physically significant fitting parameters that quantify the dynamic behavior of the reactor system. This article briefly describes the development of stochastic models under this effort[12] and then presents the results of noise diagnostic applications with the use of measured data from an operating PWR. The results obtained indicate the capabilities of this approach and provide insight into the behavior of the dynamic reactor system. Analyzing neutron noise data for diagnostic content in the manner presented in this work provides a suitable basis for automated, on-line diagnostic applications, such as plant life extension monitoring and plant performance monitoring. The use of such diagnostic capabilities can lead to improved plant safety, availability, and reliability through the detection of incipient malfunctions or identification of progressive system degradation.

STOCHASTIC MODELS OF NEUTRON NOISE

The Feedback Dynamics Model

The description of the low-frequency neutron PSD from an ex-core detector is based on a low-order model of the full primary system in a PWR. This model consists of two modules of equations representing the dynamic behavior of the reactor core and the steam generator loops. The core module is composed of a system of distributed parameter balance equations that describe the space and time dependence of the field variables. The core neutronics equations were developed with variational techniques to retain the axial dependence of the neutron and precursor populations. This derivation allows the spatial "view" of the ex-core detectors along the core height to be maintained. The neutron–thermal–hydraulic feedback is represented by axially dependent temperature feedback coefficients. The radially averaged core thermal–hydraulic equations are given by energy balances for the fuel and coolant as well as mass and momentum balances for the coolant in a representative channel. These equations were made stochastic by including disturbances of the field variables and introducing parametric fluctuations to yield a set of Langevin equations.[2] The stochastic mass and momentum equations provide models of the field variable noise sources. Also, empirical correlations between parameters and field variables were used to represent parametric fluctuations in terms of material property perturbations and field variable fluctuations. Coupling these noise source models with the stochastic neutronics and energy balances gives a set of Langevin equations (i.e., stochastic input/output equations) relating the fluctuations in neutron power, precursor population, fuel temperature, and coolant temperature to spatially uncorrelated Ornstein–Uhlenbeck processes,[13] which yield white noise sources in the temporal and spatial limits.

The steam generator module is given by a set of lumped-parameter, coupled equations describing the energy balances in the steam generator and the hot- and cold-leg piping. The steam generator primary coolant energy balance is coupled to the dynamics of the balance of plant through the tube metal energy balance. The Langevin approach was followed to yield a set of stochastic equations relating steam generator inlet and outlet temperatures, steam generator tube metal temperature, and core inlet temperature to perturbations of the coolant velocity and heat-transfer coefficient in the steam generator and to secondary steam pressure fluctuations. These noise sources are related to coolant flow noise and power demand fluctuations. The velocity and heat-transfer fluctuations were assumed to be white noise, and the pressure fluctuations are measured field variable perturbations. This module allows the propagation of effects through the closed primary loop and the coupling of the nuclear steam supply system with the energy conversion system to be taken into account in the neutron noise model. It is coupled to the core module through the boundary condition equating the core inlet temperature to the axially dependent core coolant temperature evaluated at the bottom of the core channel.

Fourier transforming the module equations led to the closed form expression for the normalized power fluctuations given by

$$\theta_p = T_c Q_c + T_s Q_s \quad (1)$$

where T_c and T_s are space and frequency dependent operators (i.e., spatially dependent transfer functions), Q_c represents the axially distributed noise sources in the core module, and Q_s gives the sources arising from the steam generator system, including the stochastic load perturbations represented by the measured secondary steam pressure fluctuations. The detector response fluctuations caused by neutron power disturbances are given in terms of the power fluctuations, the detector impulse response, and the weighted detector capture cross section. Therefore the neutron PSD from an ex-core detector depends on the squared modulus of the measuring equipment's transfer function and the product of the Fourier-transformed power fluctuations and its complex conjugate, integrated over the detector length. With the use of the closed form representation of the neutron power fluctuations, given by Eq. 1, in the relationship for the detector response, an analytical expression was obtained for the neutron PSD in terms of source magnitudes and frequency-dependent shape functions that are derived from the physical model. The functional expression of the neutron PSD allows the model to be brought into agreement with reactor noise data by adjusting the source magnitudes with parameter estimation techniques.

The Mechanical Motion Model

The model of the response of an ex-core neutron detector to small mechanical motions of reactor internals requires a mathematical description of the interaction of mechanical vibrations with the neutronic field. For the derivation of the mechanical motion model, the reactor core was partitioned into three-dimensional zones with moving boundaries driven by turbulent coolant flow forces. The movement of large internal structures, such as the core support barrel, corresponds to the collective movement of grouped zone boundaries. The ex-core detectors are at rest in zones outside the boundaries of the pressure vessel. The moving zone interfaces were represented by time-dependent boundary coordinates composed of steady-state and fluctuating components. The assumption was made that the zone boundary motions could be described by second-order systems and that the turbulent driving forces were white noise sources. Therefore the amplitudes of mechanical motions are expressed by the convolution of the random driving forces, evaluated on the interfaces at rest, with damped oscillatory contributions dependent on the material properties of the zones, which determine the stiffness and damping characteristics.

The effects of zone interface motion on the core neutronics were taken into account by perturbing the boundary conditions of the Boltzmann equation describing the neutron flux in each zone. Formulating the detector response to the neutronics of the reactor and applying perturbation theory using the flux, boundary conditions, and detector response led to an expression for the detector fluctuations in terms of scale factors for the interface motions that act as weights measuring the effect of the stochastic driving forces on the detector response. These weights or "window" functions arise from flux gradient mismatches at the unperturbed boundaries and operate on the driving forces. The window functions are convolved with the vibratory characteristics of the interface itself, as expressed by the second-order mechanical system transfer function, to give the contribution from a resonant motion to the detector signal fluctuation. Each contribution is summed to give the total perturbed detector response.

Each motion was grouped according to vibration frequency rather than zone boundary to simplify the expression of the detector fluctuations. Next, the Fourier transformed detector response and its complex conjugate were multiplied to give an expression for the neutron PSD. After some further grouping of terms, the expression for the neutron PSD reduced to

$$\Phi_{AA}(\omega) = \sum_{\lambda} \left[\frac{\mu_\lambda A_\lambda + (\omega - \nu_\lambda) B_\lambda}{\mu_\lambda^2 + (\omega - \nu_\lambda)^2} \right] + BG(\omega) \qquad (2)$$

where λ = index that varies over the frequencies of the mechanical vibrations
A_λ = pole strength or amplitude of the λth resonance
B_λ = asymmetry or skewness factor for the λth resonance
μ_λ = damping coefficient for the λth resonance
ν_λ = damped frequency of vibration for the λth resonance
BG = background arising from the thermal–hydraulic feedback dynamics

The pole strength describes the magnitude of the effect of the vibration on the detector response. The skewness factor represents the amount of interference in the detector response to one particular vibration that results from competing perturbations introduced by other vibrations.

The shapes given by the PSD model peak at the resonance frequency for each λ mode of motion. In cases where the "tails" or off-resonant vibrations of modes at different frequencies do not contribute significantly to the amplitude of a modal resonance peak (i.e., there is light modal coupling and the measurement data are predominately due to the one vibration mode), the resonance parameters of that mode can be determined by a single mode fit. However, in instances where the measurement data in the vicinity of a peak are strongly influenced by off-resonant contributions (i.e., there is heavy modal coupling and the interference of the tails is not negligible), all the modal parameters must be identified simultaneously or the interference among the different vibrational modes must be determined. Neutron power spectral densities typically exhibit heavy modal coupling in the frequency range of interest for vibrational studies. This effect is described by the asymmetry factors in the mechanical model of the PSD. By including these terms in the description of the vibrational PSD, it is possible to characterize the coupling between modes of vibration as evidenced in the detector's response to their effect on the neutron flux density, thereby separating the motions and allowing the resonance parameters for each peak to be extracted. The form of the mechanical motion model readily permits a quantitative investigation of the resonance structure of the neutron PSD through identification of four physically significant parameters for each peak.

NEUTRON NOISE DIAGNOSTIC APPLICATIONS

The stochastic models of the neutron PSD represent tools that can be used to diagnose in a systematic fashion the information on the dynamic condition of the reactor system available from ex-core detector noise data. By adjusting these models to fit reactor data, it is possible to determine physically significant parameters that quantify the dynamic behavior of the plant. In addition, the fitted models can be used to investigate how the structure of the PSD evolves in response to alterations in the state of the reactor system characterized by changes in neutronic and thermal–hydraulic parameters. A systematic diagnostic methodology using stochastic models, parameter estimation, sensitivity studies, and long-term observation and analysis for trending important dynamic indicators is the ultimate goal toward which this work is directed.

The application of the feedback dynamics model illustrates the use of stochastic models, adjusted to represent real data, to determine diagnostic information on how changes in the plant condition will be evidenced as spectral changes. Such information can be used to develop rules for detection and diagnosis that can be incorporated into expert systems. The mechanical motion model is used to analyze data taken periodically over an extended time period, covering one fuel cycle and the beginning of a second cycle. This analysis identifies trends in the vibratory behavior of the in-vessel components.

Data from ex-core power range monitors at the Tennessee Valley Authority's Sequoyah Unit 1 Nuclear Power Station, an 1148-MW(e) pressurized-water reactor of the Westinghouse four-loop design, were recorded on magnetic tape periodically from 1981 to 1983 by researchers from ORNL. The analog data recordings began shortly after the start of power operation of the unit and continued into the second fuel cycle at the plant. The recordings taken in 1983 correspond to the beginning of the second fuel cycle. The data were digitized and reduced to normalized frequency domain spectra. The applications described in this article make use of selected data recordings taken at full power and flow conditions during this time period.

System Feedback Dynamics Identification from Low-Frequency Noise

The Functional Fit. The stochastic feedback dynamics model of the low-frequency neutron PSD structure was incorporated into a generalized least-squares-fitting code to allow comparison of the model predictions with actual plant data and to permit adjustments of the noise source magnitudes. By determining the model source amplitudes through the fit to neutron detector data, the model is brought into good agreement with the noise descriptors of the nuclear plant and can be used to characterize the dynamic state of the reactor.

With the low-frequency spectrum obtained from data taken late in the first fuel cycle at the subject PWR, the least-squares adjustment program accomplished a functional fit of the feedback dynamics model to estimate model source amplitudes. The adjusted model prediction shows good agreement with the major features of the measured PSD, falling within a statistical error band of three standard deviations around the measured PSD over the full frequency range. Figure 1 shows the fit obtained in this study and illustrates that the model provides a reasonable description of the major features of the PWR noise descriptor. The error band shown in the figure is based on the Fourier analysis statistics and the variance of the reduced data.

On the basis of the determined source strength coefficients, the parametric fluctuation sources related to reactivity effects and heat-transfer coefficient perturbations (which arise from material property effects and turbulent flow conditions at the fuel assembly walls) are found to be strong sources of noise. The relatively high magnitude found for the secondary steam pressure source can be attributed to its importance at low frequencies. The use of this measured source represents an attempt to account for the unmodeled dynamics, such as long-term controller action and balance of plant dynamics.

The feedback dynamics model provides a reasonable representation of the PSD structure at very low frequencies, and it matches the spectral shape above 0.01 Hz very well. It is this frequency range from 0.01 to 1 Hz that is most affected by the characteristic core residence time, plant heat-transfer time constants, and coolant temperature and velocity fluctuations[2] that are included in the feedback dynamics model. Therefore the adjusted model

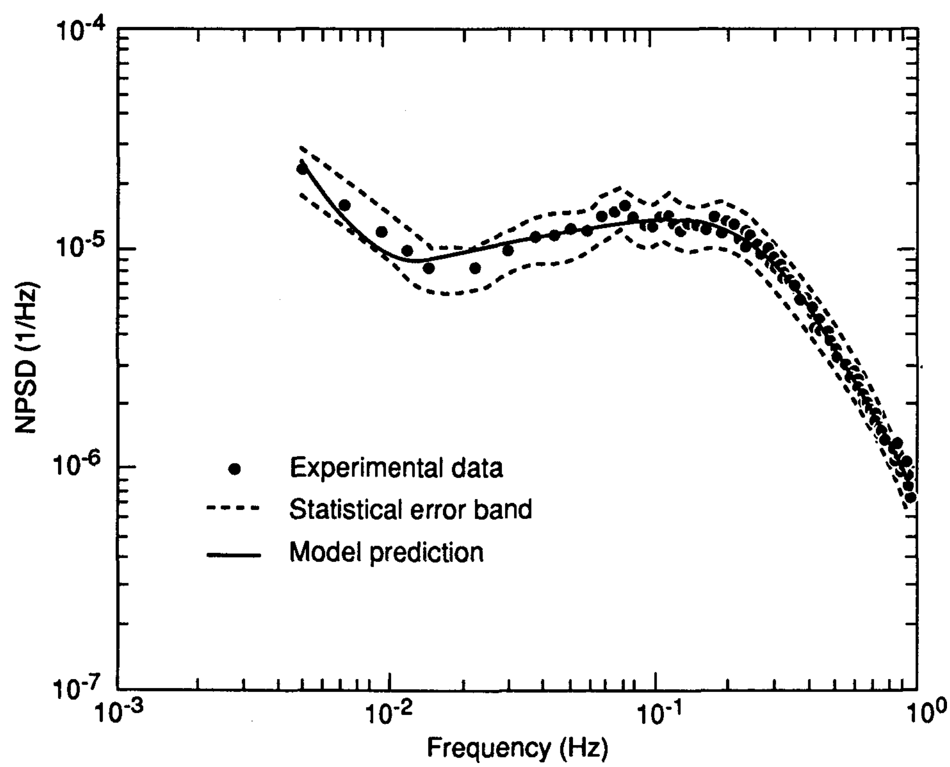

Fig. 1 Fit of feedback dynamics model to the low-frequency normalized power spectral density (NPSD) from an ex-core neutron detector.

can be used to represent the dynamic condition of the reactor at the time of the measurement. This model provides the basis for a limited diagnostic analysis of the reactor condition and its relation to certain physically significant parameters.

Diagnostic Evaluation of the Adjusted Model Predictions. *Surveillance system discriminants.* Currently, statistically based pattern recognition systems are being used for continuous, on-line surveillance of dynamic reactor signals.[14] Such automated reactor noise surveillance systems use statistical methods to compare PSD measurements with baseline PSDs from the same reactor system. Changes in the monitored reactor signals from the reference condition are detected and stored for later diagnostic analysis. For noise surveillance, a ratio is formed between the test PSD and the reference PSD for all frequency estimates. This ratio is then compared with discriminants that have been formulated to emphasize relevant features in the PSD.

Various discriminants[14] have been devised to detect fluctuations in the integral power of the spectrum, magnitude changes of the spectrum limited to narrow frequency bands, spectral shape changes, and shifts in spectral peak frequencies. In this study, two discriminants were chosen to test the detectability of changes in certain physical parameters that describe the dynamic behavior of PWR systems. Because of the large dynamic range typically associated with PSDs and the monotonic nature of logarithms, the log of the ratios of the PSDs is used for these comparisons. The first discriminant is the mean ratio determined from the set of test PSD to reference PSD ratios obtained at N individual frequencies. This measure of the integral difference between spectra is given by

$$D_I = \frac{1}{N} \sum_{i} \left\{ \log \left[\Phi_t(\omega_i) \right] - \log \left[\Phi_r(\omega_i) \right] \right\} \quad (3)$$

Because the mean is taken of the ratio of the PSDs, each frequency range of the spectrum is, in a sense, normalized so that the mean ratio discriminant gives equal weight to all PSD components. As a result, the discriminant provides a check of spectral differences over each frequency range of interest, regardless of the absolute magnitude of the spectra in that range. It is easily seen that the discriminant offers a measure of the average difference between spectra in a frequency range. Since a uniform spectral shift will cause the log of the test PSD to be either greater or smaller than the log of the reference PSD for multiple estimates, the mean ratio is sensitive to such shifts over the frequency range of interest. However, if the frequency range contains uniform shifts of opposite direction, the discriminant is subject to cancellation effects. As a result, the mean ratio discriminant is limited in its ability to detect spectral variations if offsetting deviations are present.

To overcome the cancellation limitation of the mean ratio discriminant, the second discriminant is constructed using the second moment of the log of the ratios of the PSDs. This measure of the variance of the set of ratios is given by

$$D_{II} = \frac{1}{N} \sum_{i} \left\{ \log \left[\Phi_t(\omega_i) \right] - \log \left[\Phi_r(\omega_i) \right] \right\}^2 \quad (4)$$

This discriminant deals with the average squared distance between the test PSD and the reference PSD, on a log scale, at several estimates over a frequency range. Note that, by summing and averaging the squared distances between the log test and log reference spectra, this discriminant avoids the cancellation effect of the first discriminant to give a better indication of spectral shifts, but it loses the ability to determine the direction of the shifts. As a result, it was decided to use both discriminants to characterize PSD behavior in this study.

The first discriminant is called the mean log ratio (MLR), and the second discriminant is called the log ratio variance (LRV). In surveillance systems, the discriminants used to monitor reactor signals are checked against an alert level and an alarm level to determine if a change in the spectrum has occurred. The criteria for these two levels are initially predicted on the basis of theoretical assumptions concerning the signal, and these criteria are checked and modified during the learning phase of the surveillance period. Current systems begin surveillance with the assumption that the signals have Gaussian amplitude distributions and that their individual PSD estimates are independent. From these assumptions, a theoretical confidence interval is calculated giving the width in standard deviations around the discriminant median into which its value must fall for the signal to successfully pass its test against the reference PSD. The alert confidence level lies within that set for the alarm level. As the learning phase progresses, the assumptions about the nature of the signals are tested, and the confidence intervals are updated with measured means and standard deviations.

Analysis of the spectral structure evolution. In this work the MLR and LRV discriminants were used along with the fitted parametric neutron PSD model to study the detectability of changes in various physical parameters that characterize the dynamic state of the reactor system. After a parameter value was changed by some percentage, the model PSD was recalculated and compared with the "baseline" model PSD. The means and standard deviations used to determine the normalized indicator were calculated with the number of data blocks comprising the original measured PSD to which the model had been fit. Since this was not an ongoing surveillance application, no updating of the confidence intervals occurred, and the results are limited by the assumptions concerning the nature of the original measured signal.

After obtaining adjusted model predictions that were representative of the reactor state during normal operation, a direct sensitivity study of the effects of changes in physical parameters on the major features of the neutron PSD was performed. It was found that the spectral shape of the neutron descriptor was sensitive to changes in the moderator temperature feedback coefficient (a reactivity effect), coolant residence time in the core (a core flow effect), and core heat transfer. It was also found to be sensitive to a lesser degree to changes in the steam generator heat transfer and in the thermodynamic state of the secondary steam (a load effect). Figure 2 illustrates the frequency ranges where changes in each of these parameters were evidenced by changes in the noise descriptor. In addition, indication is given on the figure of whether the relationships between changes in parameters and spectra are directly proportional (+) or inversely proportional (–). Changes in the core heat transfer affect the spectrum over two frequency ranges such that increases in the heat-transfer coefficient cause positive shifts in the spectrum at very low frequencies and at relatively high frequencies. The reactivity effect is such that increases in the magnitude of the moderator temperature feedback coefficient result in an increase over the entire frequency range of the spectrum with a more pronounced effect in the 0.1- to 1-Hz range. This result corresponds to the observations of neutron noise spectral evolution over a fuel cycle where the magnitude of the PSD increases with decreasing boron concentration (i.e., an increasingly negative moderator temperature feedback coefficient). Finally, the core flow changes cause a split effect. Below 0.1 Hz, an increase in core flow causes a negative spectral shift, whereas the same change causes an increase in the spectrum above that frequency. This results from the reduced core residence time, which

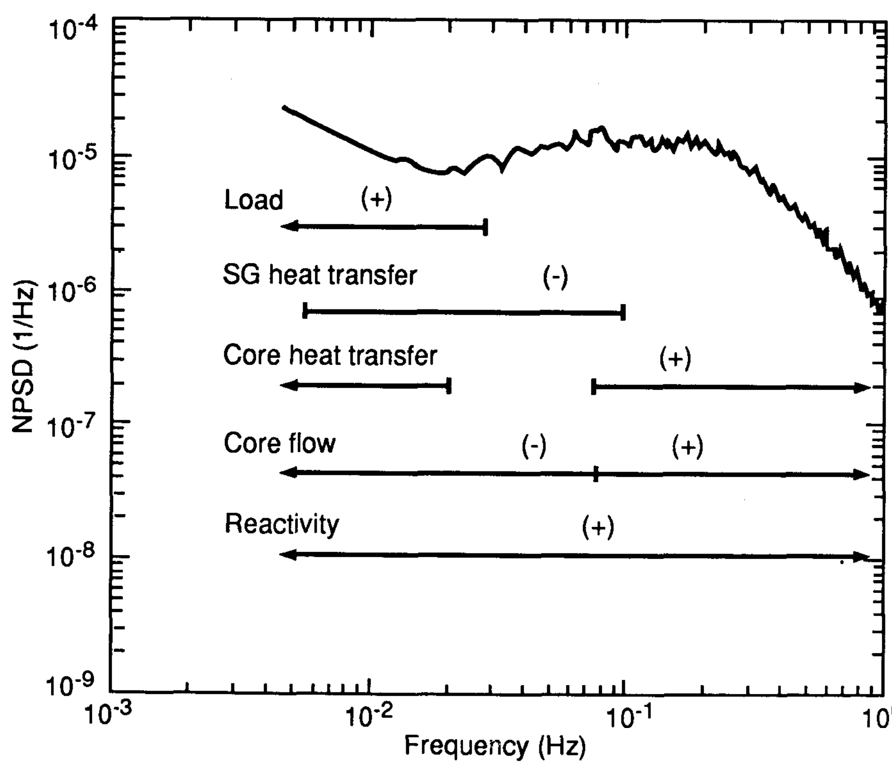

Fig. 2 Frequency bands showing sensitivity of the model to changes in physical parameters. NPSD, normalized power spectral density. SG, steam generator.

causes the flow-induced dynamic effects to occur at correspondingly higher frequencies.

Given the sensitivity information obtained, an effort was made to determine the detectability of such changes with current surveillance techniques. With the use of the model to generate a "baseline" spectrum and modified spectra for comparison, the previously described discriminants were calculated for various altered parameter sets and were checked against the initial alert level used in current surveillance systems. Thus it was possible to determine the magnitude of parametric changes that would be detected by monitoring the noise descriptors for spectral shifts. The five parameters identified as significant in the sensitivity study were used. For each parameter, the frequency range tested was chosen on the basis of the sensitivity analysis. The load variations and the steam generator heat-transfer changes were found to be detectable by only the MLR discriminant and only in extreme cases (i.e., parameter changes of 80% or greater). For core heat transfer, changes of around 10% in the heat-transfer coefficient were detectable by the MLR discriminant, and changes of above 30% caused an alert for the LRV discriminant. Changes of 10% in core flow triggered an alert for the MLR discriminant. Finally, reactivity feedback changes of 5% were detected by the MLR discriminant, and changes of 20% were detected by the LRV discriminant.

The diagnostic information on the dependence of the PSD structure on physical parameters and the detectability of changes in those parameters can be used in expert diagnostic systems in the form of monitoring and detection criteria and heuristic rules for diagnosis of observed deviations from the baseline. This type of systematic evaluation can provide insight into the behavior of the dynamic system as observed through neutron noise and illustrates the diagnostic information that can be extracted with stochastic models adjusted to represent measured noise descriptors.

Analysis of the Vibratory Behavior of PWR Internals

The Evolution of Spectral Resonances. The neutron PSD from an ex-core detector at a PWR is characterized by resonances in the 1- to 20-Hz frequency range. The major sources of these resonances are vibrations of the pressure vessel and the internal mechanical structures of the core. In the 1- to 10-Hz range, the sources of the PSD resonance structure are dominated by fuel assembly and core support barrel motion effects. Thermal shield, pressure vessel, and higher order motions of internals provide the major influence on neutron noise in the 10- to 20-Hz range. Beam mode vibrations characterize the lower frequency resonances, whereas shell mode vibrations occur in the higher frequency range.

Over the course of a fuel cycle, the structure of the neutron PSD in the vibration resonance frequency range changes as components "age" and the core neutronics change as the result of differing boron concentrations. Figure 3 shows the variation of the PSD structure over time (given in effective full-power days), and the resonances are attributed to the motion of particular components (e.g., pendular motion of the core support barrel at 6 to 7 Hz and thermal shield shell mode vibration at ~12 Hz). The amplitude of the PSD increases as the fuel cycle progresses and then is reduced to nearly the same level at the beginning of the next fuel cycle. This increase in the noise signal is attributable for the most part to fuel burnup and decreasing boron concentration, which increases the scale factor for detection of vibrations causing flux perturbations.[3] It has been postulated that the noise does not return to the same level because the clamping of the core support barrel at the beginning of the first fuel cycle and the stiffness of the fuel assemblies in the full core at the beginning of its life lead to reduced amplitudes of vibration for the new core.[3]

As shown in Fig. 3, the frequencies of the fuel assemblies and the core support barrel decreased during the fuel cycle. The beginning-of-life fuel assemblies have as much as a 10% greater natural frequency than at the end of life because of a decrease in stiffness.[15] The second cycle core contained old and new assemblies, so the frequency shift is moderate and the amplitude of vibration is greater than the start of the first fuel cycle. The shift in the core barrel peak probably results from a relaxation in clamping force during the cycle.[3] The reinstallation of the pressure vessel head at the start of the second cycle tightens the clamping and causes a slightly greater vibration resonance frequency for those measurements.

The ex-core detector sees composite peaks composed of many resonances at like frequencies. Some of these peaks have been related to classes of motion for particular types of internals. Figure 3 shows the combination of the two distinct resonance peaks in the 5- to 9-Hz range into a single peak at about 8 Hz. This important effect results because the resonant peaks from the core support barrel and the second mode of fuel assembly vibration shift until they are close in frequency while increasing in amplitude so that they become visually inseparable as the fuel cycle progresses. Therefore it becomes difficult to isolate changes to the core support barrel clamping with-

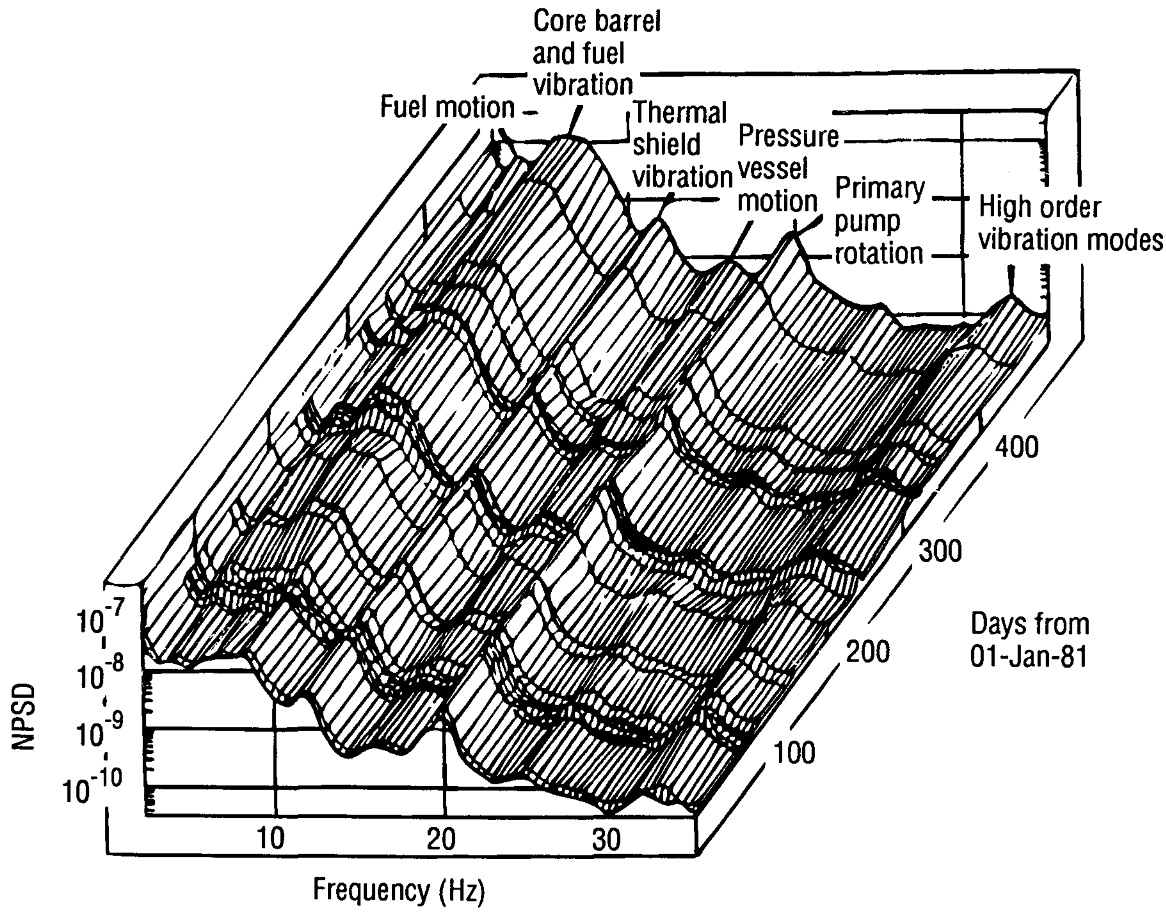

Fig. 3 Evolution of the normalized power spectral density (NPSD) from neutron noise over the first and second fuel cycles at a pressurized-water reactor.

out a means to separate the motions. This represents a major consideration in the development of the mechanical motion model and is addressed in the discussion of the application of that model. Also, other peaks arise during the fuel cycles. For example, the resonance at 4 Hz shows up as a distinct peak during the second fuel cycle, although it begins to emerge late in the first fuel cycle. The source of this resonance is undetermined, but it may result from vibration of fuel assemblies with different stiffness properties than that of most of the elements visible to the detector. The effect of submerged peaks can be accounted for by adding fitting peaks to represent them.

Parameter Estimation Over a Fuel Cycle

The mechanical motion model was implemented as user-supplied function and derivative subroutines in the generalized least-squares-fitting code. For this application, the low-frequency representation generated by the fitted feedback dynamics model was used as the background term. The background parameter included in the mechanical motion model fit represents the integral magnitude of the feedback dynamics contribution to the spectra.

The frequency range chosen for this application was limited to 14 Hz and below. Task size limitations are the main reason for this choice. It was determined that seven peaks were needed to describe the data available from the first and second fuel cycles. The four major peaks are the first mode of fuel assembly vibration at 3 to 3.5 Hz, the core support barrel vibration at 6 to 7 Hz, the second mode of fuel assembly vibration at 7 to 8 Hz, and the thermal shield vibration at 11.5 to 12 Hz. The additional fitting peaks are less distinct and have not been attributed to particular components. Indeed, the "peaks" at 2 and 9 Hz are more accurately described as "bumps" on the spectra, whereas the 4-Hz peak is visible as a distinct peak only in the data from late in the first fuel cycle and in the second fuel cycle. It may be that this peak and the

9-Hz feature arise from fuel assemblies whose stiffness remains higher than that evidenced by the majority of the elements visible to the detector (i.e, the hold-down springs do not "relax" as much as for most of the core, and the natural frequency remains higher). For the second fuel cycle, this effect would result from new assemblies placed in the outer positions of the core. It is important for trending purposes that comparisons be made between parameters determined from like models, so the use of seven peaks was maintained throughout this application.

The values of the resonance parameters for each fit following the first were used as initial parameter guesses for the subsequent fit. In this way the insight into the evolution of the spectra gained at each application of parameter estimation was used as an *a priori* input to the next fit. This proved valuable in the cases where the core support barrel and second fuel mode vibrations were closely coupled, and a visual estimation of starting frequencies and amplitudes would have been difficult.

Figure 4 provides an example of the results obtained by showing a selected measured PSD and its associated model fit. The agreement between the fitted shape and measured data demonstrates the ability of the model to provide an excellent representation of the PSD. Resonance parameters determined for the four major peaks using data recorded during the first and second fuel cycles are given in Table 1. The data recordings listed for 1983 are from the second fuel cycle.

Trending Vibration Peak Evolution and Separating Motions

As discussed previously, the core support barrel and second mode of fuel assembly vibration merge into what appears to be one effective peak as the fuel cycle progresses. As a result, vibration monitoring systems that do not account for the coupling between resonances in the detector's "view" may have difficulty isolating the behavior of the individual peaks. In the mechanical motion model developed for this work, such interference in the way a detector sees each peak is taken into account. Figures 5 and 6 show the spectral decomposition of the model predictions for the beginning of the first fuel cycle and the beginning of the second fuel cycle. These plots illustrate the asymmetry occurring in the spectral resonance contributions of the core support barrel and second fuel assembly vibration mode as the peaks combine in the PSD. By fitting over the fuel cycle, the evolution of this effect can be monitored, and mistaken diagnosis concerning the core barrel vibratory behavior can be avoided.

Certain general trends for the amplitudes and resonant frequencies of the four major resonances through the first

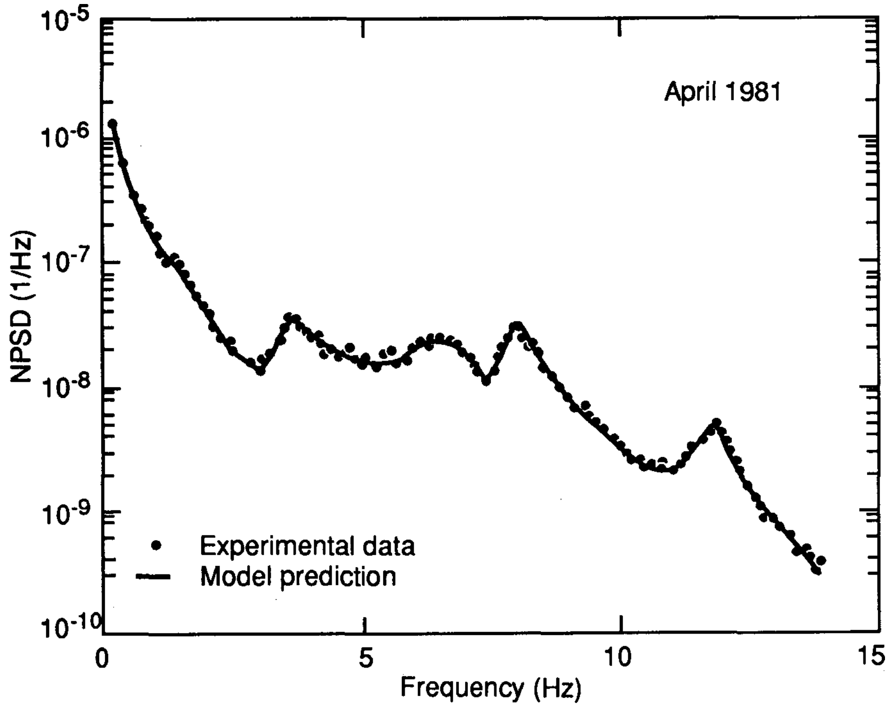

Fig. 4 Fit of the mechanical motion model to the high-frequency normalized power spectral density (NPSD) from neutron noise.

Table 1 Mechanical Motion Model Parameters for the Major Vibrational Peaks Selected Recording Times[a]

Date recorded	First fuel mode		Core barrel		Second fuel mode		Thermal shield	
	A_λ	ν_λ	A_λ	ν_λ	A_λ	ν_λ	A_λ	ν_λ
April 1981	4.32×10^{-8}	3.56	1.09×10^{-7}	6.53	5.20×10^{-8}	7.95	1.11×10^{-11}	11.72
January 1982	5.47×10^{-8}	3.37	5.40×10^{-7}	6.06	2.82×10^{-7}	7.62	1.18×10^{-11}	11.90
April 1982	6.99×10^{-8}	3.41	6.26×10^{-7}	6.10	2.81×10^{-7}	7.74	1.50×10^{-11}	11.87
June 1982	6.05×10^{-8}	3.36	6.59×10^{-7}	6.00	3.33×10^{-7}	7.56	1.25×10^{-11}	11.85
August 1982	7.45×10^{-8}	3.16	6.68×10^{-7}	5.76	7.21×10^{-7}	7.31	1.42×10^{-11}	11.79
March 1983	9.98×10^{-8}	3.04	2.22×10^{-7}	6.99	1.26×10^{-7}	8.10	1.17×10^{-11}	11.94
April 1983	1.05×10^{-7}	3.00	3.06×10^{-7}	6.59	2.91×10^{-7}	7.41	1.69×10^{-11}	11.82
August 1983	2.77×10^{-7}	2.96	3.66×10^{-7}	5.99	9.72×10^{-7}	7.42	2.95×10^{-11}	11.88

[a] A_λ represents the normalized resonance amplitude, and ν_λ gives the vibrational frequency in Hz.

fuel cycle and into the second fuel cycle were demonstrated by the fitted parameters. As expected, the amplitudes increase over time as the soluble poison concentration decreases. Also, the fuel assembly vibration and core support barrel peaks show a decrease in the vibration frequency as the mechanical constraints of the components relax with time.

The information gained from this application to measured data supports the observations made by previous researchers[3] on the vibratory behavior of the internals at this plant. The use of this technique in an automated system would permit the trending of the resonance parameters and a comparison with expected or previously discerned trends. In addition, this model provides the

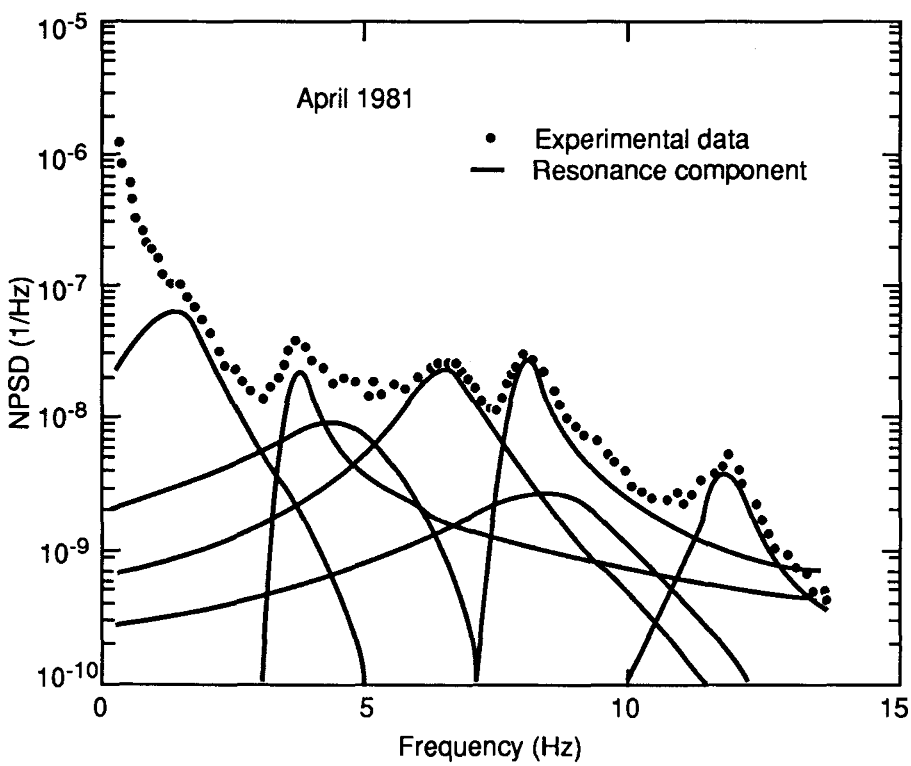

Fig. 5 Decomposed model prediction showing separated resonance contributions (first fuel cycle). NPSD, normalized power spectral density.

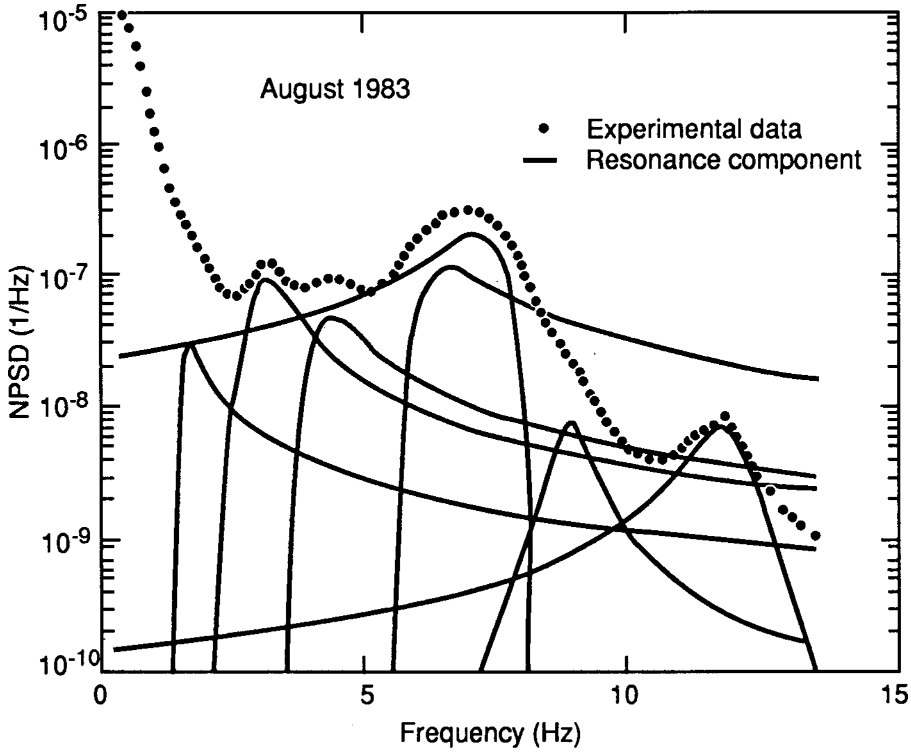

Fig. 6 Decomposed model prediction showing separated resonance contributions (second fuel cycle). NPSD, normalized power spectral density.

capability to separate the effect of the motions on the neutron PSD and thus isolate key resonances for monitoring. Since it is important to compare fitted parameters from comparable models, the automated system can be configured to fit the model with a varying number of peaks at each analysis point to provide a set of resonance parameters that can be used should unanticipated "subterranean" peaks emerge or visible peaks disappear. By application of this systematic parameter identification technique as part of a long-term monitoring and diagnostic system, insight into the vibratory condition of the reactor internals can be gained.

CONCLUSIONS

Two stochastic models describing the ex-core neutron PSD have been developed at ORNL. The feedback dynamics model describes the neutronic–thermal–hydraulic feedback dynamics that are dominant in the low-frequency range of the neutron PSD. Axially dependent balance equations in the core and lumped parameter energy equations for the rest of the primary coolant circuit form the basis for the model. The low-frequency neutron PSD is expressed in terms of frequency-dependent operators derived from the dynamic balance equations and in terms of stochastic noise sources arising from the core and steam generator system. The mechanical motion model was developed from perturbation theory to give the detector response to small in-core mechanical motions. The motions are characterized in the model by resonance parameters that can be determined by a functional fit to measured data. In addition to the customary vibration frequency, damping, and peak amplitude parameters, this model includes a skewness factor that represents the effect each resonance has on the detector's view of other vibration resonances in the core. This allows the motions to be separated when there is heavy coupling between peaks that may appear as a single peak in the spectra.

These models were incorporated into a fitting code and adjusted with measured data from the high- and low-frequency ranges so that they represented the observed dynamic state of the reactor system. The determined parameters and resulting representations of the PSD were then evaluated for diagnostic content. The use of the feedback dynamic model allowed the behavior of the

PSD in response to changes in physical parameters to be evaluated by a direct sensitivity analysis. By coupling this study with surveillance discriminants from an automated monitoring system developed at ORNL, the threshold for detection of selected parameter variations was determined, and the frequency range over which these spectral indicators are significant was determined. Thus the effect of such variations in the reactor condition on observable features in neutron noise descriptors was investigated. By evaluating the stochastic model predictions after adjustment to match measured noise data, greater insight into the nature of the relationship between the structure of neutron PSDs and physical parameters describing the system has been gained.

The mechanical motion model was used to quantify resonance peaks in neutron PSDs taken over a fuel cycle. By monitoring the evolution of the spectral peaks over time, it was possible to trend the change in vibratory response of selected structures within the core. Of particular note was the observation of the relaxation of the stiffness of the core's mechanical configuration, detected by a shift in the fuel-element vibration frequencies, and the ability of the model to separate the resonance peak corresponding to core support barrel pendular motion from the peak indicative of the second mode of fuel vibration when the two peaks merged to form a single broad spectral peak as the fuel cycle progressed. The use of this model to quantify resonance peaks and trend the vibratory behavior of the monitored internals over a fuel cycle represents a viable technique for automated surveillance and analysis of the structural integrity of the in-vessel components. The inclusion of the skewness factor and the ability of the technique to separate the spectral effects of the motions enhances the potential for its use as a diagnostic tool.

These applications demonstrate the capability of using stochastic modeling as an aid to understanding the complicated information retrieved from power reactor noise measurements. The information obtained from such analyses can be incorporated into surveillance systems to focus application of the detection discriminants to allow important physical parameters to be monitored or to trend important parameters allowing for maintenance scheduling or incipient failure detection. In addition, including this information in expert diagnostic systems can allow the dynamic condition of the neutronic, thermal–hydraulic, or mechanical behavior of the plant to be diagnosed from spectra that deviate from the baseline. Early detection of undesirable operating conditions or degraded mechanical components by using such surveillance and diagnostic techniques in expert advisors for operations and maintenance would enhance plant safety. In addition, the ability to automatically, nonintrusively monitor the structural integrity of reactor internals and the dynamic condition of the reactor system can provide additional safeguards for plants that are granted life extension.

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Design Features

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Westinghouse Advanced Passive 600 Plant

By B. A. McIntyre[a] and R. K. Beck[a]

Abstract: *The Westinghouse advanced passive AP600 nuclear power plant has been designed to meet the needs for electrical power generation into the 21st century. This article discusses the innovative features of the AP600, including the nuclear steam supply system, the passive safety systems, radionuclide attenuation, the balance-of-plant design, plant arrangement, and modular construction. The AP600 test program, developed to demonstrate the functionality of the unique design features, is briefly addressed.*

For a high degree of public safety and licensing certainty, the AP600 design certification program has been structured for final design approval and design certification under 10 CFR Part 52. Additionally, the AP600 is designed to optimize established technology to ensure that there is no requirement for significant new development and no need for a demonstration plant.

Growth in electrical demand continues to be strong, but orders for new generating capacity have not kept pace. Recent orders have been primarily for gas-turbine plants to be used to offset peak power demands. Utilities will have to order new plants within the next few years to avoid a widespread shortage. Additionally, there is a growing concern about the environmental and economic risks involved with increasing dependence on large-scale fossil-fuel alternatives: oil, gas, and coal. As a result, there is a growing realization that nuclear power must play a major role in our energy future.

One of the primary impediments to new nuclear plant orders has been the uncertainty of the plant being licensed for operation once construction has been completed. Utilities are not willing to commit several billion dollars if the plant may not be allowed to operate after construction is completed. There is also a need to cut costs by constructing the plant in significantly less time than the decade or more it has taken for present-day plant construction. The new standardization regulations adopted by the Nuclear Regulatory Commission (NRC) are a great step forward. Preapproval of the site will address the emergency planning issues before plant construction. The NRC's certification of the plant design before construction begins minimizes the chance that any last-minute design issues will prevent operation once construction is finished.

To match the need for electrical power generation with the current economic and environmental constraints, Westinghouse has developed the AP600, a design for a simplified passive advanced light-water-reactor (ALWR) nuclear power plant of 600-MW(e) nominal output power rating. The AP600 design reduces cost and improves safety by incorporating predominantly passive safety systems, a simplified reactor coolant system (RCS), digital instrumentation and control (I&C), and other innovative features in an optimized plant arrangement. The application of modular construction methods and an efficient construction plan provides a significantly reduced construction time that, compared with many recent domestic nuclear projects, reduces construction costs.

The development of the AP600 began in 1985 with the Electric Power Research Institute (EPRI) Small Plant Study. This continued under the Department of Energy (DOE) Technology Programs in Support of ALWRs, which has resulted in a total plant conceptual design.

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In October 1989 DOE awarded Westinghouse a \$50 000 000 contract to further develop the AP600 conceptual design into an NRC Certified Design by the end of 1994. EPRI is also participating in this Design Certification program.

Westinghouse has applied over 30 years of experience in commercial nuclear power to all aspects of the AP600 design (including safety, licensability, manufacturability, constructibility, operability, and maintainability) to design a simpler nuclear power plant that offers many advantages over existing light-water reactors (LWRs). Experience in such areas as balance of plant design, operability, and constructibility is provided by Bechtel Power Corporation, Burns and Roe, Avondale Industries, CBI Services, MK-Ferguson, and Southern Electric International.

The innovative features, such as a simplified RCS, passive safety systems, digital I&C, optimized plant arrangement, and modular construction methods, which characterize the AP600 design, were developed to meet the following top-level plant objectives defined in the DOE and ALWR programs:

• Provide a greatly simplified plant in terms of the number of systems and equipment, operations, inspections, maintenance, and Quality Assurance (QA) requirements.

• Provide increased plant operating reliability by simplifying the plant systems.

• Provide a high degree of public safety and licensing confidence by:

—Addressing current licensing issues

—Reducing core-melt frequency to less than 1×10^{-5}/year for external and internal events

—Reducing the public risk for severe accidents

—Providing low radiological releases that support eliminating the need for evacuation beyond the plant boundary

• Ensure that a plant prototype will not be required by using experience-based power-generation system arrangements and components.

• Reduce the cost of power so that it is competitive with other power-generation options.

• Provide a short construction schedule that can be met with high confidence.

• Minimize the impact on the environment, especially with respect to:

—Heat discharge

—Radiation releases

—Chemical releases

—Solid waste removal and disposal

—Occupational radiation exposure

Table 1 provides a comparison of selected AP600 design features with a standard two-loop nuclear power plant of the same power rating. The simplified loop configuration, passive safety system, and simplified plant arrangement contribute to overall plant simplification in a complementary fashion that significantly reduces the complexity of the plant.

Table 1 Major Plant Features Comparison

Plant features[a]	Reference 600-MW(e) plant	AP600	Reduction, %
Pumps			
Safety	25	0	100
Nonnuclear safety	188	139	26
HVAC			
Fans	52	27	48
Filter units	16	7	56
Valves			
NSSS	512	215	58
BOP > 5 cm (2 in.)	2 041	1 530	25
Pipe			
NSSS	44 300 ft	11 042 ft	75
BOP > 5 cm (2 in.)	97 000 ft	67 000 ft	31
Evaporators	2	0	100
Diesel generators	2 (safety grade)	1 (nonnuclear safety)	50
Building volume			
Seismic (including containment)	9.4×10^6 ft^3	4.6×10^6 ft^3	51
Nonseismic	6.2×10^6 ft^3	6.1×10^6 ft^3	2

[a]HVAC, heating, ventilation, and air conditioning; NSSS, nuclear steam supply system; BOP, balance of plant.

AP600 NUCLEAR STEAM SUPPLY DESCRIPTION

The RCS illustrated in Fig. 1 uses two cold legs and one hot-leg pipe per loop. Two canned-motor pumps are close-coupled to each steam generator. A larger pressurizer is attached to one of the hot legs.

The core design uses a low-power-density reactor core consisting of 145 fuel assemblies of the 17×17 Vantage 5 design with an active fuel length of 3.7 m (12 ft). The design is based on well-developed, low-enriched fuel core technology. Soluble boron and burnable poisons are used for shutdown and fuel burnup reactivity control. Low worth grey rods are included for load follow and

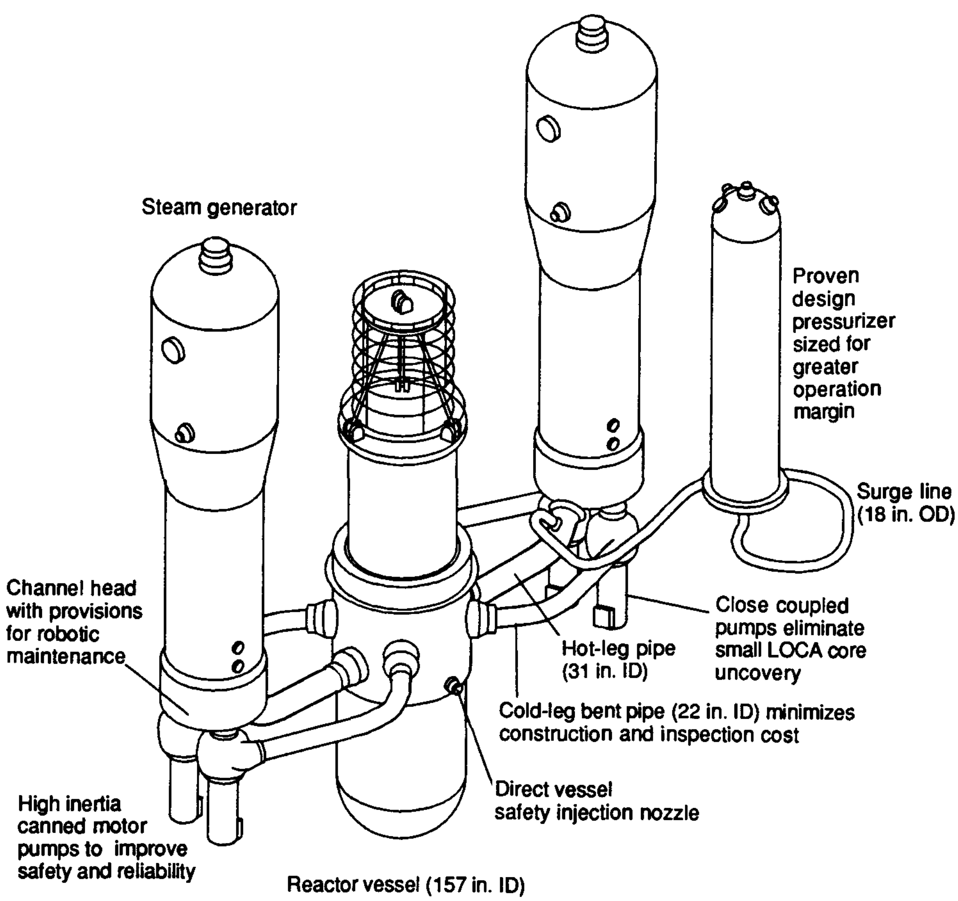

Fig. 1 AP600 reactor coolant system. LOCA, loss-of-coolant accident.

power regulation. A reference fuel cycle of 18 months with a three-region core has been selected to enhance plant availability. The core is surrounded by a stainless steel and water neutron radial reflector, which serves to reduce neutron leakage, enrichment, and fuel-cycle cost. The reactor vessel and internals are of essentially conventional design, so no manufacturing development is required.

The steam generator consists of a Westinghouse Model F secondary side (shell, tube supports, baffles, separators, dryers, and feedwater header) and a primary-side channel head modified to permit the direct attachment of two canned-motor reactor coolant pumps. The plant layout provides ample access and space for tube inspection, plugging, and sleeving by either manual or robotic means while preserving good nozzle entry and exit flow characteristics. A robotically handled multiport nozzle dam permits steam generator inspection and maintenance operations while reactor refueling operations are under way. Sufficient space is available within the containment to allow steam generator replacement.

The hermetically sealed canned-motor pumps are mounted in the inverted position and close-coupled to the steam generator. The pump suction nozzles are welded to vertical channel head outlet nozzles, which effectively combine the steam generator and reactor coolant pumps into a single structure and eliminate the need for a separate set of pump supports.

The canned-motor reactor coolant pump was selected because of its demonstrated record of high reliability and simplified auxiliary fluid systems. Since the canned motor in this pump eliminates the shaft seal, safety is enhanced by eliminating the possibility of a shaft seal loss-of-coolant accident (LOCA). In addition, the loop seal is eliminated by this design. This eliminates core uncovery during a postulated small-break LOCA.

A heavy uranium disk has been incorporated (canned) within the motor to increase the rotating inertia of the canned-motor pump. The increased rotating inertia improves the performance of the pump during loss-of-flow transients.

The I&C architecture is based on a unified design approach with distributed, digital microprocessor-based technology with electrical and fibre-optic data links. The integrated I&C systems include the plant protection system, the nuclear and balance-of-plant control system, the operational display system, the alarm system, the accident monitoring system, the plant computer, the control board, and the emergency control board. Other equipment, such as radiation monitoring, metal impact, flux mapping, and failed fuel detection, share a common plant parameter data base with the integrated systems via the monitor bus.

Westinghouse has experience with distributed digital systems in nuclear applications from work done in this country with the EAGLE-21 control and protection systems upgrades. The recently awarded contract for the Sizewell B control and protection system will demonstrate the software and hardware technology for the AP600 systems.

The main control board and the emergency control board designs take advantage of the microprocessor-based I&C equipment. Standardized push-button control stations are used to communicate with the protection and control systems via electrical data links. Power for the I&C equipment required for reactor trip and accident mitigation is provided by Class 1E batteries.

Human factors are being considered throughout the I&C and control room design to enhance operability and to decrease the probability of operator error. This includes task analyses of the various operator functions, development of human performance models, and the design of the soft controls for the operator's workstations.

The CRTs and qualified plasma displays provide the information the operator requires for normal and emergency operation of the plant. An advanced alarm system categorizes, prioritizes, and displays alarm messages and suppresses minor "nuisance" alarms.

PASSIVE SAFETY SYSTEMS

The AP600 uses passive safety systems to enhance the safety of the plant and to satisfy NRC safety criteria. These systems use only natural forces, such as gravity, natural circulation, and compressed gas, to make the system work. No pumps, fans, diesels, chillers, or other rotating machinery are used. A few simple valves, as described later in this section, are used to align the passive safety systems when they are automatically actuated. In most cases these valves are "fail safe" [i.e., they require power (normally supplied by Class 1E uninterruptible power supplies) to stay in their normal, closed position; loss of that power causes them to open to their safety alignment]. The passive safety systems are significantly simpler than typical pressurized-water-reactor (PWR) safety systems.

These passive safety systems have no pumps and 37 remote valves, compared with 6 pumps and over 100 remote valves in a typical active safety injection system. In addition, these passive safety systems are comparable with traditional design for the rest of the plant, core, RCS, and containment. The changes made to the balance of the plant optimize performance and incorporate lessons learned from experience that will make the AP600 easier to maintain. Probabilistic risk assessment (PRA) has been used as a part of the design process to optimize such items as valve arrangement, valve type, and electrical configuration.

In addition to being simpler, the passive safety systems do not require the large network of safety support systems needed in typical nuclear plants—such as a-c power; heating, ventilation, and air conditioning (HVAC); and cooling water systems—and seismic buildings to house these components. This simplification includes eliminating the safety-grade emergency diesel generators and their network of support systems, air start, fuel storage tanks and transfer pumps, and the air intake–exhaust system. As a result, these support systems no longer need to be safety grade and can be simplified or eliminated. For example, the essential service water system and its associated safety cooling towers are eliminated. Figures 2 and 3 compare the AP600 heat removal systems with those of a current PWR and illustrate how much simpler the AP600 systems are.

The features of the AP600 passive safety systems include passive safety injection, passive residual heat removal, passive containment cooling, and passive control room habitability under emergency conditions. All these passive systems have been designed to meet the NRC single-failure criteria. PRAs have also been used to quantify the safety of the design. The passive system designs address recent NRC criteria, including Three Mile Island Nuclear Station (TMI) lessons learned, unresolved safety issues, and generic safety issues.

The readiness of these systems is determined by monitoring the tank levels and by periodic cycling of the few alignment valves. In addition, during plant shutdown all

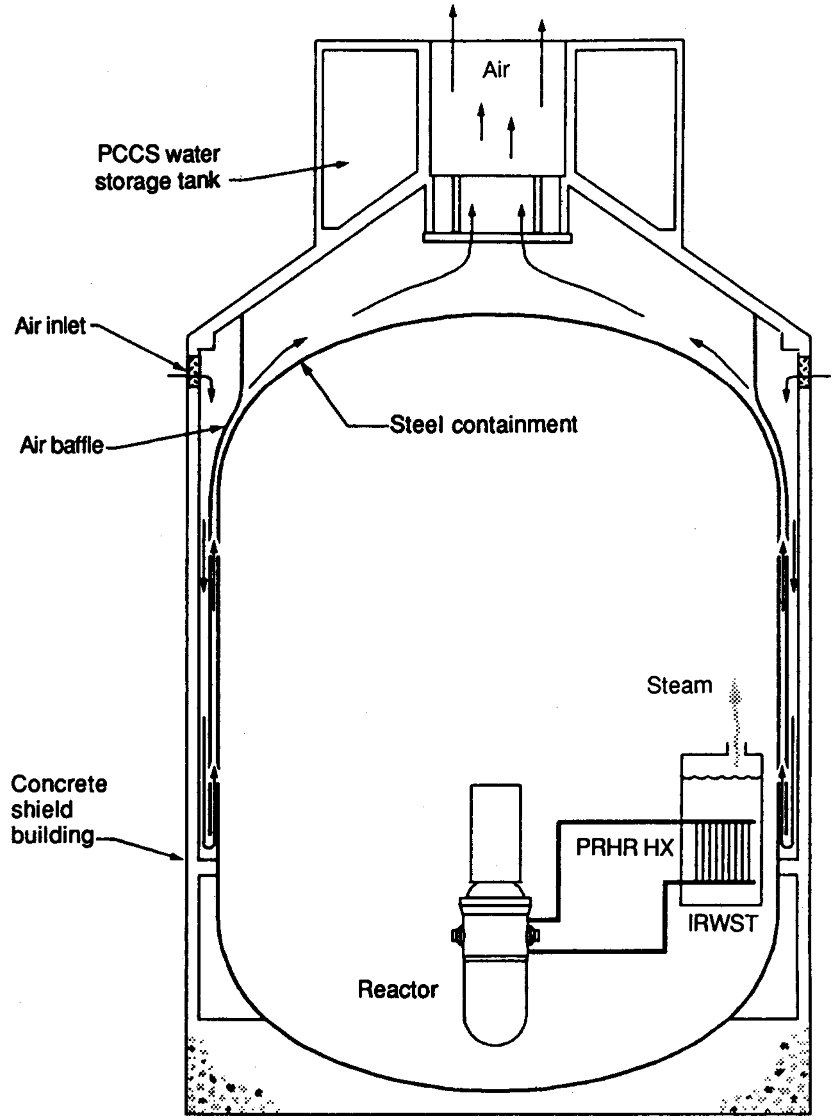

Fig. 2 AP600 plant heat sink. PCCS, passive containment cooling system; PRHR HX, passive residual heat removal heat exchanger; IRWST, in-containment refueling water storage tank.

the passive features will be tested to demonstrate flow and heat-removal performance.

Several aspects of the passive safety systems have been used in existing nuclear plants. The accumulators are a part of most PWR designs, so their use is well understood. Several early boiling-water reactors (BWRs) used isolation condensers as natural-circulation closed-loop heat removal systems. The passive residual heat removal heat exchanger (PRHR HX) was designed with the benefit of this experience. The use of isolation on the outlet side of the PRHR HX instead of on both sides will prevent water hammer during actuation. Boiling-water reactors have used automatic depressurization systems (ADS) and spargers for many years. The use of slow opening valves is a result of understanding the air-clearing loads experienced in BWR operation.

Passive Safety Injection System

The passive safety injection system (PSIS) (Fig. 4) performs three major functions: residual heat removal, reactor coolant makeup for inventory control, and safety injection. Preliminary results of computer analyses with NRC-approved codes demonstrate that the PSIS provides

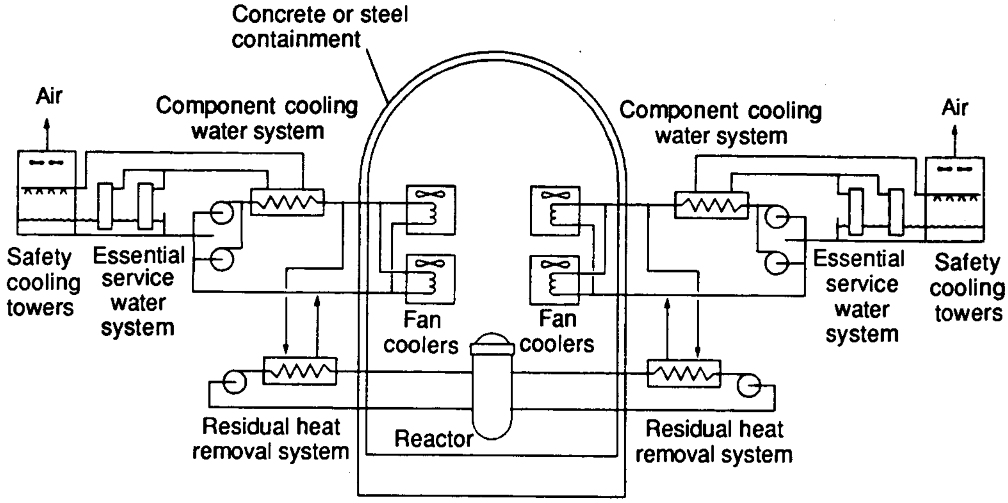

Fig. 3 Conventional pressurized-water-reactor plant heat sink.

effective core cooling for various break sizes and locations. These calculations show that the PSIS prevents core damage for breaks as large as the 20-cm (8-in.) vessel injection lines and provides about 371°C (700°F) margin to the maximum peak clad temperature limit for the double-ended rupture of a main reactor coolant pipe.

The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. Preliminary analysis results, with NRC-approved codes, have shown it to satisfy the NRC safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks with a single failure. Anticipated transients without reactor trip have also been analyzed and shown to result in peak RCS pressures of about 20 MPa(g) (2900 psig), which is well within NRC criteria. The PRHR HX consists of two 100% capacity banks of tubes connected to the RCS in a natural circulation loop. The loop is normally isolated from the RCS by two 25-cm (10-in.) air-operated ball valves. These valves are normally closed and fail open if power is lost. The heat exchanger tubes are located in the in-containment refueling water storage tank (IRWST). This location places the PRHR HX above the RCS loop such that hot water leaving the RCS hot leg will rise to the top of the PRHR HX where it is cooled. The difference in temperature between the hot inlet water and the cold outlet water drives the natural circulation loop. The PRHR natural circulation will be confirmed as part of the startup testing program. If the reactor coolant pumps are running, they boost the PRHR HX flow.

The IRWST provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for about 2 hours before the water starts to boil. After that time steam that would enter the containment would be generated. This steam would condense on the steel containment vessel and then drain back into the IRWST. Although no operator action is required to keep the cooldown rates within limits, the operator is provided the capability of controlling the PRHR HX flow rate so that he or she can control the RCS cooldown if desired.

The PSIS uses three sources of water to maintain core cooling, including core makeup tanks (CMTs), accumulators, and the IRWST. All these injection sources are connected directly to two nozzles on the reactor vessel. These connections, which have been used on existing two-loop plants, reduce the possibility of spilling part of the injection flow.

Passive reactor coolant makeup is provided to accommodate small leaks following transients or whenever the normal makeup system is unavailable. Two core makeup tanks (CMTs), filled with borated water, are designed to provide this function at any RCS pressure using only gravity as a motive force. These tanks are designed for full RCS pressure and are located above the RCS loop piping. If the water level in the pressurizer reaches a low-low level, the reactor is tripped, the reactor coolant pumps are tripped, and the CMT discharge isolation valves open automatically. The relative elevations of the CMTs and the pressurizer are such that if RCS level continued to decrease the water in the CMTs would drain into the reactor vessel.

For the accommodation of large leakage rates, including LOCAs up to the postulated double-ended break of a main loop pipe, initial safety injection is provided by the

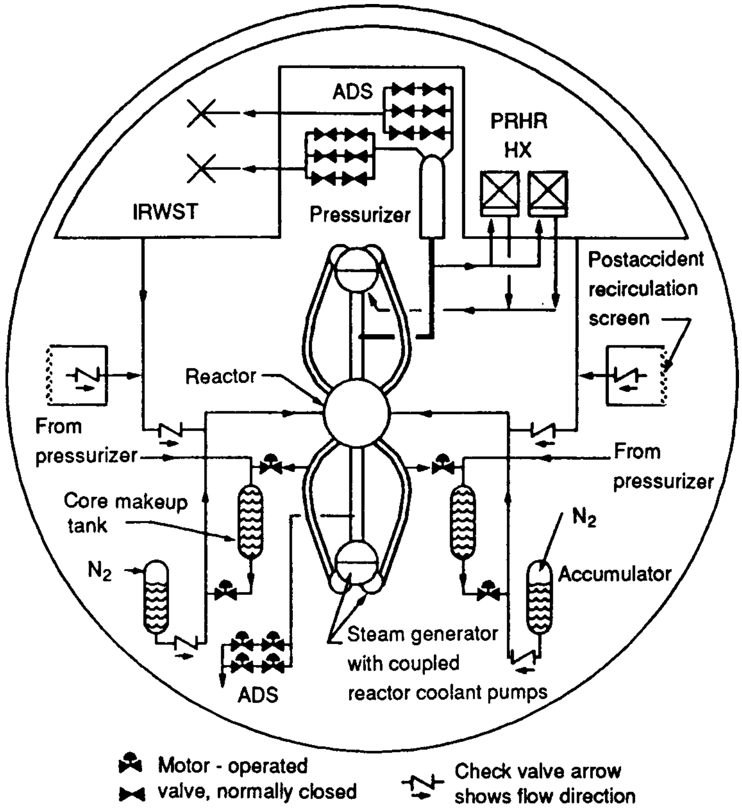

Fig. 4 Schematic representation of the in-containment passive safety injection system. IRWST, in-containment refueling water storage tank; PRHR HX, passive residual heat removal heat exchanger; ADS, automatic depressurization system (four stages).

two CMTs described previously and two accumulators. One CMT and one accumulator use a common injection line directly to the reactor vessel downcomer, so at least one CMT and accumulator are available following a postulated injection line break. A pressure balance line is connected from an RCS cold leg to the top of each CMT. This line permits a large amount of steam to flow to the top of the CMT, which results in a high flow rate of water to the RCS.

As with current PWRs, accumulators are required for large LOCAs to meet the need for higher initial makeup flows to refill the reactor vessel lower plenum and downcomer following RCS blowdown. Each of the accumulator tanks contains 48 m^3 (1700 ft^3) of borated water with about 8.5 m^3 (300 ft^3) of nitrogen at 4.8 MPa (700 psi). The gas-pressure forces open check valves that normally isolate the accumulators from the RCS. The accumulators are sized to respond to the complete severance of the largest RCS pipe by rapidly refilling the vessel downcomer and lower plenum. The accumulators continue delivery to assist the CMTs in rapidly reflooding the core.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by self-actuating check valves. This tank is designed for atmospheric pressure. As a result, the RCS must be depressurized before injection can occur. The AP600 automatically controls depressurization of the RCS to reduce its pressure to about 69 kPa(g) (10 psig), at which point the head of water in the IRWST is sufficient to overcome the small RCS pressure and the pressure loss in the injection lines. The automatic depressurization system (ADS) is made up of four stages of valves to permit a relatively slow, controlled RCS pressure reduction. The first three stages are connected to the pressurizer and discharge through spargers into the IRWST. The fourth stage is connected to a hot leg and discharges through redundant isolation valves to the containment. All the ADS stages are actuated by CMT level. All the valves make use of existing nuclear-grade valve body and operator designs.

During a LOCA the IRWST will provide injection for at least 10 hours. As the IRWST level approaches empty, the containment water level will exceed the RCS loop level. This level is sufficient to force water to drain through a screen and check valves back into the RCS where it will be turned into steam by core decay heat. The steam will be vented to the containment through the ADS valves and the break, where it will be condensed on the inside of the steel containment vessel. The condensation will drain down into the lower part of the containment and become available for injection into the RCS again.

The passive containment cooling system (PCCS) provides the safety-related ultimate heat sink for the plant (see Fig. 2). As demonstrated in the preliminary computer analysis and tests, the passive containment cooling is capable of effectively cooling the containment following an accident such that the design pressure is not exceeded and the pressure is rapidly reduced. The steel containment vessel itself provides the heat-transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Steel containment vessels of similar size have been used on 15 operating PWRs. Heat is removed from the containment vessel by a natural circulation flow of air that cannot be isolated. During an accident the air cooling is supplemented by evaporation of water on the outside of the containment shell. The water is drained by gravity from a tank located on top of the containment shield building. Two normally closed, fail-open butterfly valves are opened to initiate the water drain. The water tank is sized for 3 days of operation, after which time the tank is expected to be refilled so that the low containment pressure achieved after the accident can be maintained. If the water is not resupplied after 3 days, the containment pressure will increase; however, the peak is calculated to reach only 90% of design pressure after about 2 weeks.

The PCCS is required to perform its containment heat removal function only when the normal means of containment heat removal (the containment fan coolers) are unavailable for an extended period or following a postulated design-basis event that results in a large energy release into the containment.

Westinghouse preliminary analyses show that the AP600 has a significantly reduced frequency of release of large amounts of radioactivity following a severe accident core-melt scenario. This analysis shows that, with only the normal air cooling, the containment stays well below the predicted failure pressure. Other design factors contributing to this result include improved containment isolation and reduced potential for LOCAs outside containment. The improved containment performance makes it technically feasible to eliminate the emergency planning zone.

Radionuclide Attenuation

A goal of the AP600 program is to limit the whole-body dose to less than the 1-rem (0.01-Sv) limit at the site boundary. This will support the technical justification to reduce the emergency planning zone to the 0.5-mile (0.8-km) site boundary. Several design features were investigated to provide, in conjunction with a revised source term, sufficient radionuclide attenuation to meet this goal.

The containment leak rate is sufficiently low that, if credit is taken for settling of particulate matter in the auxiliary building, there is no need for any other mechanical systems to address radionuclide attenuation. The AP600 containment penetrations are located so that any leakage is into the auxiliary building.

Since there is no need for a containment spray system for the purpose of radionuclide attenuation or to reduce the containment pressure following an accident, there is no containment spray system in the AP600.

AP600 BALANCE OF PLANT DESIGN

The AP600 plant arrangement consists of six principal building structures, each constructed on individual flat basemats that provide seismic, cost, and construction schedule advantages. They are as follows:

- Nuclear island
- Turbine island
- Annex building
- Diesel generator building
- Solid radwaste building
- Access control building

Building volumes have been minimized in each of these areas without compromising any of the plant layout criteria or equipment maintainability. This reduced volume provides a direct means for shortening the construction schedule and lowering the plant capital cost. Equipment modules fabricated offsite and shipped by rail will be inserted into preconstructed areas of the plant. Several areas of the plant will be under construction at the same time, and thus the overall construction schedule will be shortened.

Plant Arrangement and Construction

The modular construction approach and simplified design will allow a 5-year lead time from plant order to commercial operation. This lead time consists of an 18-month preconstruction period for site preparation, site-specific engineering and procurement of long lead items; a 36-month construction period from first structural concrete to fuel load; and a 6-month startup and test period.

The overall plant layout provides a relatively small nuclear island with good construction access to all buildings and supports the 36-month construction schedule. The layout has a single personnel entry point for maximum security, separate "clean" and "potentially contaminated" access corridors, and areas well-defined to control personnel radiation exposure levels consistent with as low as reasonably achievable.

All safety-related systems are located within the containment building, auxiliary building, and fuel-handling building, all of which are on a common basemat. During normal operation, all reactor coolant is kept within the containment building by locating the chemical and volume control system in a shielded, prefabricated module within the building. This has resulted in a major reduction in the amount of shielding needed outside containment. The IRWST is located below the operating floor. The steel containment is surrounded by the passive, natural-convection cooling system located within the shield building.

The equipment modules, fabricated offsite and shipped by rail to the plant site, will be joined together into larger modules at the site and inserted into preconstructed areas of the plant. Several areas of the plant will be under construction concurrently. A combination of conventional concrete and rebar construction methods, prefabricated steel modules assembled onsite and filled with concrete after placement in the building, and precast concrete will optimize construction schedule and cost factors.

SAFETY AND LICENSING

One of the major objectives of the AP600 plant development program is to provide a high degree of public safety and licensing certainty. Realization of this objective within the framework of existing NRC regulations has been given high priority throughout the AP600 design and development effort. The AP600 is being designed to meet the criteria of the ALWR Passive Plant Utility Requirements Document. This will provide that the significant regulatory issues will have been resolved on a generic basis before the AP600 is submitted for NRC review in 1992.

The work completed to date has established the overall design configuration of the AP600 as well as its licensability. The AP600 Detailed Design and Design Certification Program has been structured so that:

- Final design is approved (FDA).
- Design certification is per NRC rules.
- No significant new development effort is required.
- No demonstration plant is required.
- High confidence in meeting licensing schedules is maintained.

The success of the AP600 in meeting safety objectives is confirmed by the results of transient and accident analyses as well as a core-melt frequency evaluation.

Preliminary transient and accident analyses were performed to demonstrate that the passive safety systems are effective in mitigating the consequences of design basis events.

Key results are as follows:

- *Loss of all normal feedwater.* The PRHR HX removes sufficient heat and prevents the need for operation of the pressurizer safety valves.
- *Steam generator feedline break.* The PRHR HX removes sufficient heat and prevents the need for operation of the pressurizer safety valves.
- *Steamline breaks.* No fuel departure from nucleate boiling occurs, and there is no adverse effect on the primary system.
- *Inadvertent depressurization system actuation.* The core remains covered, and transition to the long-term cooling mode is accomplished.
- *Small-break LOCA.* The core remains fully covered for breaks up to and including one of the 20.3-cm (8-in.) CMT/accumulator injection lines.
- *Large-break LOCA.* The limiting large-break peak clad temperature is 788°C (1450°F), which is significantly below the 1204°C (2200°F) Appendix K limit.
- *Limiting containment response.* Containment pressure peaks at 44 lb in.$^{-2}$ following the blowdown of a large-break LOCA and is decreased to less than 0.15 MPa (22 psi) after 1 day.
- *Offsite dose analysis.* Releases are well within the NRC limits for design basis core releases to containment.

A preliminary core-melt frequency analysis was performed for internal events. A specific model was developed that included the following AP600 features: the canned reactor coolant pump, the passive safety systems, the passive containment cooling systems, and the relevant control grade active systems.

The results, shown in Table 2, indicate that the AP600 core damage frequency is at least a factor of 10 lower than the best current U.S. plants. In addition, a significant release or containment rupture has been shown to be around 100 times less likely. A final PRA, including external events, will be included with the AP600 submittal in June 1992.

AP600 TEST PROGRAM

The application of passive systems to a nuclear power plant can result in some new phenomena that must be modeled by the safety analysis methods. Some ranges of parameters must be expanded to be encompassed by methods presently in use. In addition, tests may be required to demonstrate the functionality of the unique design features. A test program for the AP600 is under way to demonstrate the passive safeguards system concepts and to document their performance characteristics.

The need for these tests was determined by examining the various accident analysis phenomena to determine where there were sufficient data available to validate the computer codes used in the analysis. Where data were not available, test programs were developed to obtain the data. The safety system test program consists of a series of single effects tests. The results of these tests will be used to develop models for the safety analysis computer codes. The computer codes will be used to integrate the various models developed from the test program.

The design of the PCCS requires that several phenomena be studied in detail. Wind-tunnel tests of a scale model of the containment-shield building structure and air inlet–exhaust locations will be performed to demonstrate adequate natural circulation air flow under all conditions. The objective is that the final design be "wind neutral," which means that the flow of air through the PCCS be neither assisted nor hindered by the direction or velocity of the wind outside the containment. A one-sixth scale model test of the PCCS air-flow path was conducted to establish the actual flow resistance for use in the computer models.

A test of the PCCS will be performed to demonstrate combined heat transfer from inside to outside containment. This one-eighth-scale test will examine the natural convection and steam condensation on the interior of the containment as well as the exterior water-film evaporation, air cooling heat removal, and water-film behavior. The effects of noncondensibles and air cooling without the exterior water film will also be examined. The test results will be compared with a detailed analytical model to verify the computer code used in the safety analysis of the containment performance.

The performance of the PCCS depends on the ability of the water film to uniformly wet the surface of the containment. A full-scale model of the water distributor

Table 2 Calculated Core Damage Frequency

Initiating event	Core damage frequency (per year): Conventional plant	Core damage frequency (per year): AP600	Reduction factor achieved by AP600
Small loss-of-coolant accident (LOCA)	8.0×10^{-6}	2.3×10^{-8}	348
Medium LOCA	5.0×10^{-6}	1.2×10^{-8}	416
Large LOCA	8.0×10^{-7}	1.5×10^{-8}	53
Transients	1.3×10^{-5}	6.4×10^{-8}	203
Loss of offsite power	6.6×10^{-6}	2.1×10^{-9}	3143
Steam generator tube rupture	1.7×10^{-6}	1.0×10^{-8}	170
Vessel failure	3.0×10^{-7}	3.0×10^{-8}	10
Anticipated transient without trip	2.2×10^{-6}	4.5×10^{-8}	49
Loss of auxiliary cooling	1.1×10^{-5}	0	Eliminated
Interfacing LOCA	1.0×10^{-6}	0	Eliminated
Total core damage frequency	5.0×10^{-5}	3.3×10^{-7}	
Total reduction factor			152

will be tested at prototypical flows to demonstrate that the mechanical design provides the proper flow distribution. The effect of dome construction tolerances will also be studied.

Full-scale tests of the automatic depressurization system valves and spargers are planned to confirm the operability of the valves and the design of the sparger and to determine the effects on the suppression tank. The results of these tests will be used in the safety analysis and structural design computer codes.

The gravity drain behavior of the core makeup tank and the operation of the tank level instrumentation that controls the automatic depressurization system actuation will be studied in a one-sixth-scale facility. The test results will be compared with a detailed condensation analysis model to verify the computer codes used for accident analysis calculations.

The PRHR HX heat-transfer capability will be examined to develop a heat-transfer correlation for use in the accident analysis computer codes and also to optimize the design of the heat exchanger.

The use of gravity instead of pumps to provide the flows in the safety systems will result in lower forces being available to open check valves. Tests will be performed on actual check valves under typical temperatures and pressures to verify the differential pressure required to open the valve disk and to maintain the disk in a full-open position.

Air-flow tests will investigate possible effects on reactor coolant pump performance caused by nonuniform channel head flow distribution. The pressure losses in the channel head nozzle dams will also be determined. Water model tests in the same facility will be used to help determine the flow, head, and efficiency data for predicting the performance of the full-scale reactor coolant pump.

A full-scale prototype of the rotor and supporting journal bearing for the canned-motor reactor coolant pump was designed and tested, including use of the depleted uranium for the necessary mass. The test verified the manufacturability of the bearing and journal and measured the friction and drag losses. This information will be used in the safety analysis computer codes.

The AP600 ADS provides that no situations occur where the plant will be at a high system pressure during an accident. The major part of any accident will occur at low system pressure. For the study of that phase of an accident, a small-scale, low-pressure test of the passive RHR system will be conducted to demonstrate the operation of the long-term gravity makeup path from the IRWST and the RCS. This test will include modeling the reactor vessel, pressurizer, primary coolant loops, lower containment structure, and the IRWST.

SUMMARY

The Westinghouse AP600 is a demonstrably simplified PWR plant that incorporates predominantly passive safety systems, a simplified reactor coolant system, digital instrumentation and control, and other innovative features in an optimized plant arrangement. This greatly simplified 600-MW(e) PWR plant has major enhancements in safety, operational reliability, licensing certainty, cycle cost, and construction schedule, when compared with existing nuclear plants.

The Westinghouse AP600 provides the NRC, through the design-certification process, with confidence that the plant is designed safely. The design certification provides the utilities with confidence the plant will be allowed to operate once construction is completed in accordance with the inspections, tests, analysis, and acceptance criteria.

Westinghouse will be submitting the Standard Safety Analysis Report and other application materials to the NRC in June of 1992 to apply for design certification of the AP600 under 10 CFR Part 52. It is our goal to have a certified design by December of 1994. With the one-step licensing and the greatly reduced construction schedule, it is possible to have an AP600 on line in the United States by the turn of the century.

System 80+™ PWR Safety Design

By C. W. Bagnal,[a] R. A. Matzie,[a] and R. S. Turk[a]

Abstract: *Since 1985, ABB Combustion Engineering Nuclear Power (CENP) and Duke Engineering & Services, Inc. have been developing the next generation of pressurized-water-reactor (PWR) plants for worldwide deployment. The goal is to make available a prelicensed, standardized plant design that can satisfy the need for a reliable and economic supply of electricity for residential, commercial, and industrial use. Safety is the hallmark. For such a design to be available when needed, it must be based on proven technology and established licensing criteria. These requirements dictate development of nuclear technology that is advanced, yet evolutionary in nature. This has been achieved with the System 80+ Standard Plant Design.*

In 1985, ABB Combustion Engineering Nuclear Power (ABB-CENP) and Duke Engineering and Services, Inc. (DE&S) joined forces under the aegis of the Electric Power Research Institute (EPRI) Advanced Light-Water Reactor (ALWR) Program to develop, in conjunction with utilities, the design requirements for the next generation of nuclear power plants.[1–9] With support from the U.S. Department of Energy (DOE), ABB-CENP and DE&S again teamed the following year to design and license System 80+, an advanced pressurized-water reactor (PWR) that meets these utility requirements.[10–13] The final version of the EPRI ALWR Utility Requirements Document was submitted to the U.S. Nuclear Regulatory Commission (NRC) in September 1990 (Ref. 14), and in May 1991 the complete 18-volume final safety analysis report for the System 80+ Standard Plant Design was officially docketed at NRC (Ref. 15). Final design approval by the NRC staff is expected in 1993, and design certification by the Commissioners is expected 1 year later.[16–19]

SAFETY REQUIREMENTS FOR ALWRs

The ALWR requirements for evolutionary plants are integral to the System 80+ design, which directly conforms to over 99% of the more than 5500 requirements applicable to PWRs. Of those requirements related to safety, which is the fundamental criterion, seven major requirements characterize the ALWR: thermal margin $\geq$ 15%; slower response to upset conditions; core damage frequency of 10^{-5} per reactor year; no fuel damage up to a 15-cm break; 8-hour station blackout coping time; large release frequency of 10^{-6} per reactor year; and large, rugged containment. The degree of extra margin built into the EPRI requirements is illustrated by the risk targets that are set an order of magnitude better than the NRC Safety Goal. System 80+ meets or exceeds all these requirements.[20]

DESIGN FOR OPERATING SAFETY

A key feature of the System 80+ Standard Design is that it represents a complete power plant: nuclear island, turbine island, and balance of plant—all integrated for safe and reliable operation.[21] Public and occupational health and safety are fully addressed; radioactive waste generation and radiation exposure are reduced during normal operation,[22] as are the probability and consequences of postulated accidents.

Integrated Safety Systems

Improved safety is a principal tenet of the System 80+ Standard Design.[23] Redundancy and diversity are the keys to a prudent balance between accident prevention and mitigation. The safety injection system (SIS) and emergency feedwater system (EFWS) are dedicated four-train systems (Figs. 1 and 2). Containment spray and safety injection pumps take water from a storage tank inside containment [in-containment water storage tank (IRWST)] and thus eliminate the need to switch from an external source and provide a semiclosed system with continuous recirculation. Emergency core coolant flows directly into the reactor vessel to provide a simpler, more reliable system that avoids the need for orificing and valve adjustments and the potential for valve misalignment inherent in cold-leg injection schemes. The containment spray system (CSS) and shutdown cooling system (SCS) are integrated, and pumps are interchangeable;

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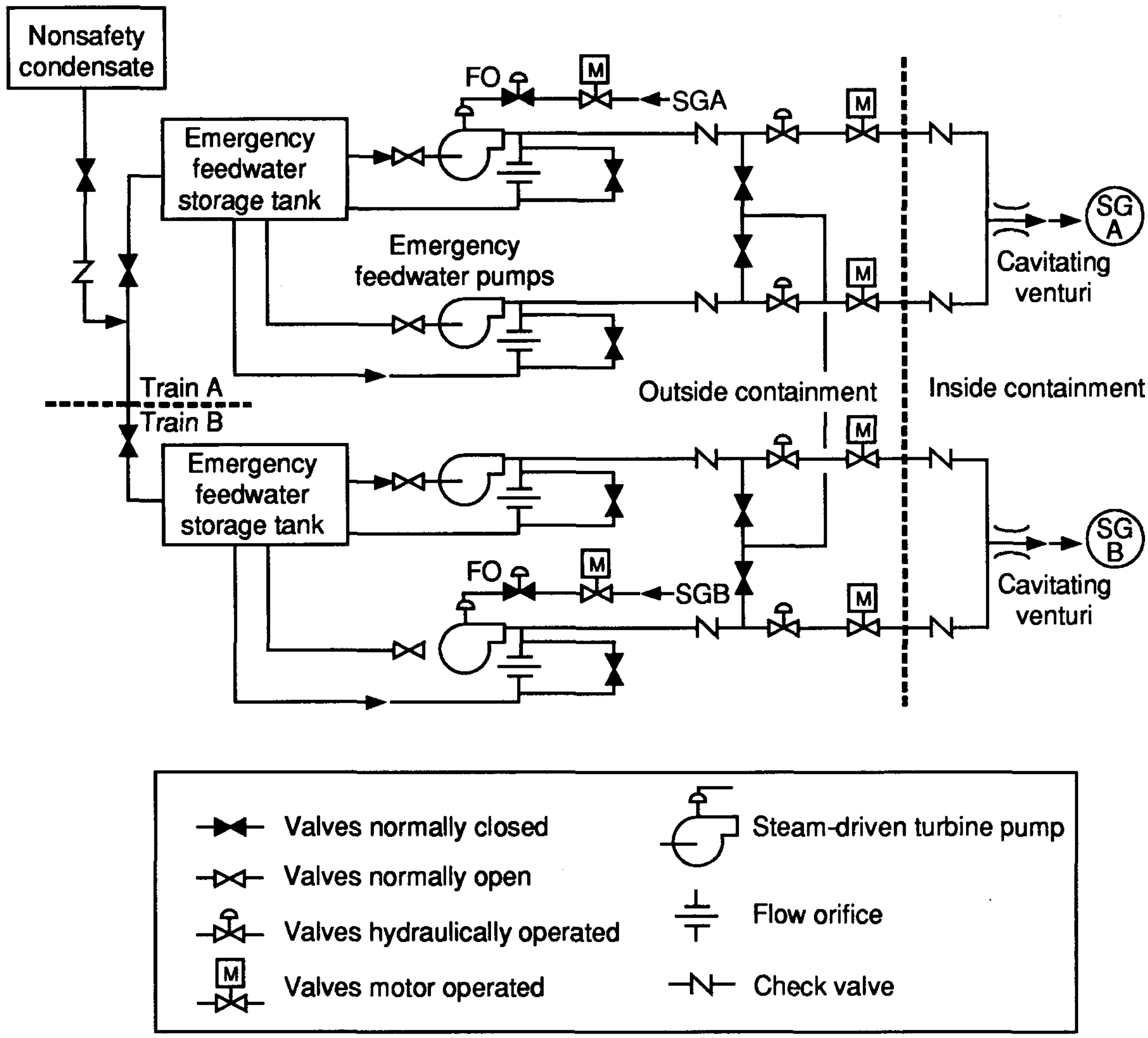

Fig. 1 Emergency feedwater system. SGA and SGB are steam generators A and B, respectively; FO is fails open.

thus backup and higher reliability are provided for both systems. Shutdown cooling is designed to maintain piping integrity even if accidentally exposed to full primary system pressure; this precludes the large interfacing system loss-of-coolant accident (LOCA) with fluid loss outside primary containment that contributed significantly to radioactive release to the environment in previous risk evaluations. Cavitating venturis minimize excess emergency feedwater flow to a steam generator with broken feed or steam lines and thus eliminate the need for automatic isolation.

Containment

The System 80+ containment structure is a major safety feature that typifies standards applied throughout the plant to make it simpler and more focused on the operator.[24–26] A spherical steel containment provides 75% more space on the operating floor than does a typical cylindrical containment of equal volume. Allowance is made for one-piece steam generator removal. Ventilation duct is modest and simple; full advantage is taken of the natural boundary of the sphere. Safety systems are located in the secure subsphere below, and thus pipe and cable are shortened from a closer nuclear annex. The subsphere allows dedicated piping and electrical areas as well as 360° access to the primary containment (Fig. 3). Quadrant division and physical separation of safety components virtually eliminate concerns of fire, flood, and sabotage (Fig. 4). A cylindrical, concrete shield building provides the added protection of a dual containment. The entire nuclear island is founded on a common basemat

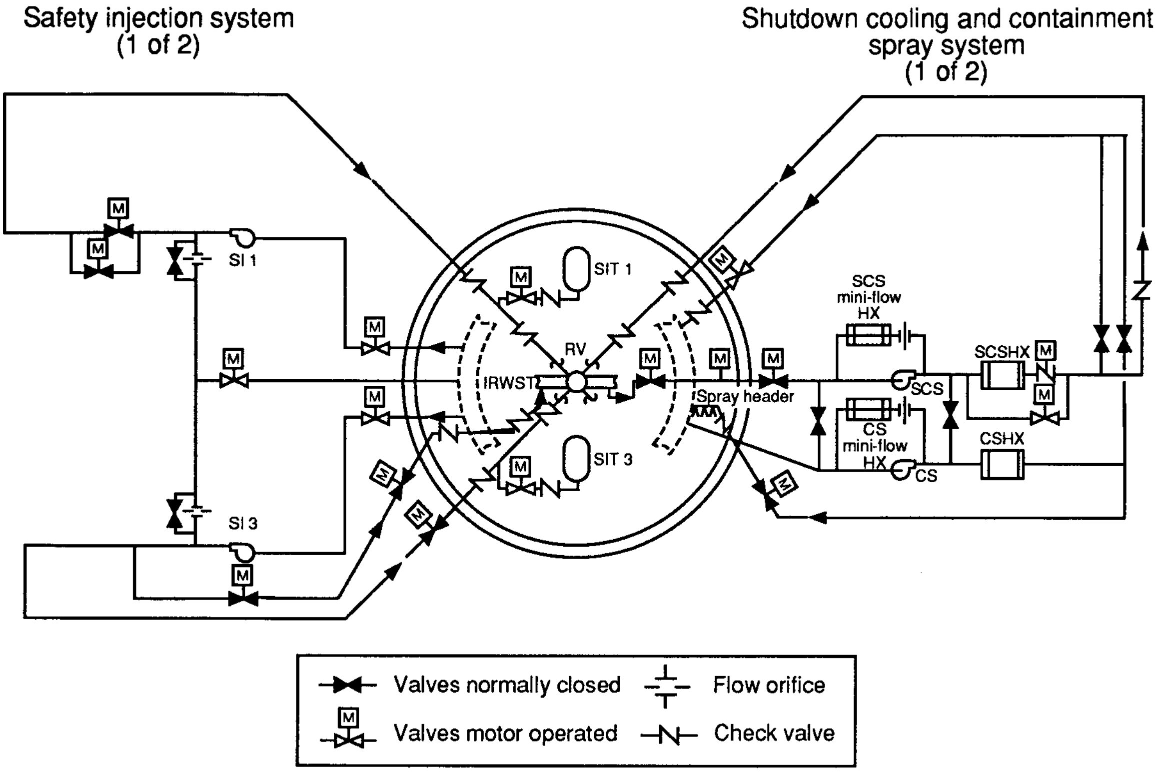

Fig. 2 Integrated engineered safety features system. SI, safety injection; SIT, safety injection tank; SCS, shutdown cooling system; CS, containment spray; HX, heat exchager; IRWST, in-containment refueling water storage tank.

embedded 16 m in the soil to provide seismic resistance.[27] The inherent strength and structural stability of the sphere preclude the need for stiffeners and postweld heat treatment.

Nuclear Steam Supply System

Increased margin and improved reliability are the keys to understanding the nuclear steam supply system (NSSS) (Fig. 5). Core outlet temperature is lowered 3.3°C from the current System 80™ design, and the reactor protective system is optimized to provide a minimum of 15% overpower margin throughout the 60-year plant life. Additional margin gains involve the use of advanced integral burnable absorbers incorporating gadolinia or erbia admixed with urania.[28] Grey rods reduce radwaste by enabling load follow without changing dissolved boron. The reactor vessel is ring-forged and manufactured with low initial RT_{NDT} material to preclude brittle fracture and reduce in-service inspection requirements by 30%. Control rod, pump seal and steam generator tubing materials, and system design ensure long life and reduce component activation and coolant contaminants. The pressurizer is 33% larger, and the steam generators include 25% more secondary inventory to slow transient response and provide margin to trip setpoints. In the event of total loss of normal feedwater flow, the initiation of emergency feed is not required for more than 30 minutes to prevent steam generator dryout. An improved steam generator dryer design provides 99.9% quality steam; heat-transfer area is increased 17%, including a 10% margin for potential tube plugging. Other steam generator improvements that promote long-term integrity, ease maintenance, and reduce personnel radiation exposure include larger and repositioned access ports, a standby recirculation nozzle and associated piping to ensure uniform chemistry during wet layup, and a redesigned flow distribution for the economizer to minimize the potential for tube vibration.

Turbine Plant

Margin and reliability define a turbine generator that provides a net power of 1300 MW(e) from a single set (Fig. 6). Full-load rejection is accepted without reactor or turbine trip. A generator output breaker permits offsite

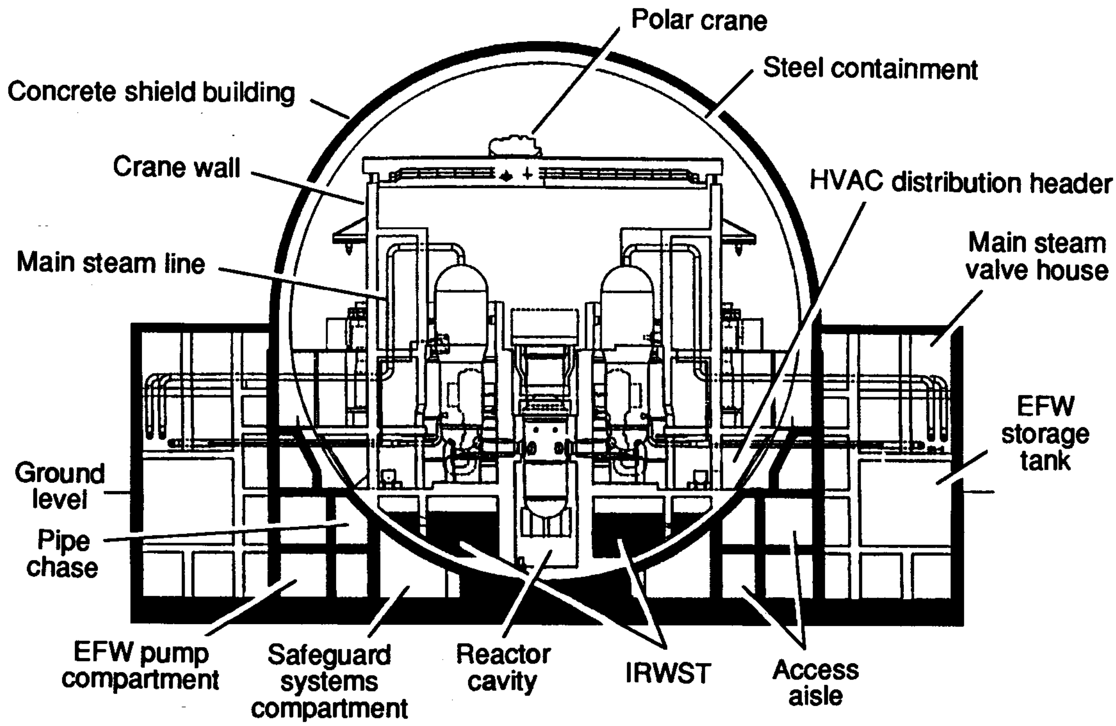

Fig. 3 System 80+ spherical steel containment. EFW, emergency feedwater; IRWST, in-containment refueling water storage tank; HVAC, heating, ventilation, and air conditioning.

supply to unit transformers during startup or emergency operation. Welded-rotor construction is used throughout to simplify manufacture and inspection and to avoid the operating risks associated with disks shrunk on the rotor shaft. A single bearing between cylinders shortens the turbine and lowers vibration. A titanium condenser significantly reduces the likelihood of corrosion products in the secondary system. A single-steam space is provided for turbine exhaust; cooling water supply and discharge pipes up to the seal pit need not be interconnected to maximize plant output on pump outages. The turbine plant is completely automatic and is supervised from the central control room.

Nuplex 80+™ Advanced Control Complex

High technology focuses the control complex on the human operator.[29,30] Programmable logic controllers, distributed system architecture, super minicomputers, fiber-optic communication, touch-sensitive cathode-ray tubes, plasma and electroluminescent displays, and advanced signal-validation techniques greatly simplify the design. Indicators are reduced 80%, 70% of conventional cabling is eliminated, and there are 60% fewer alarms for the operator to handle (Fig. 7). Diverse, spatially dedicated alarms and indicators are integrated into a dynamic information hierarchy to support the operator's needs and eliminate "backup-only" and "emergency-only" panel devices.[31] A digital plant protection system uses automatic on-line functional testing to eliminate most periodic surveillance tests.[32,33] A large plant-overview screen, visible from anywhere in the control room, provides integrated plant status at a glance, including key variables, high-priority alarms, and critical functions. Nuisance alarms are completely eliminated with mode-dependent indication (Fig. 8). Licensed reactor operators remain an integral part of a multidisciplinary design team that also includes nuclear systems engineers, human factors specialists, and instrumentation and control engineers.

Results

Conventional safety analysis shows marked improvement over System 80 (Refs. 34 to 37). Peak pressure is reduced more than 7 bar for feedwater line break. For small-break LOCAs, peak fuel-cladding temperature is reduced more than 300°C (Fig. 9). With four-train, direct-to-vessel injection, the core is under 1.1 m of water at all times, even for a 25-cm-diameter cold-leg break; for System 80, the core is exposed (Fig. 10). There is no return

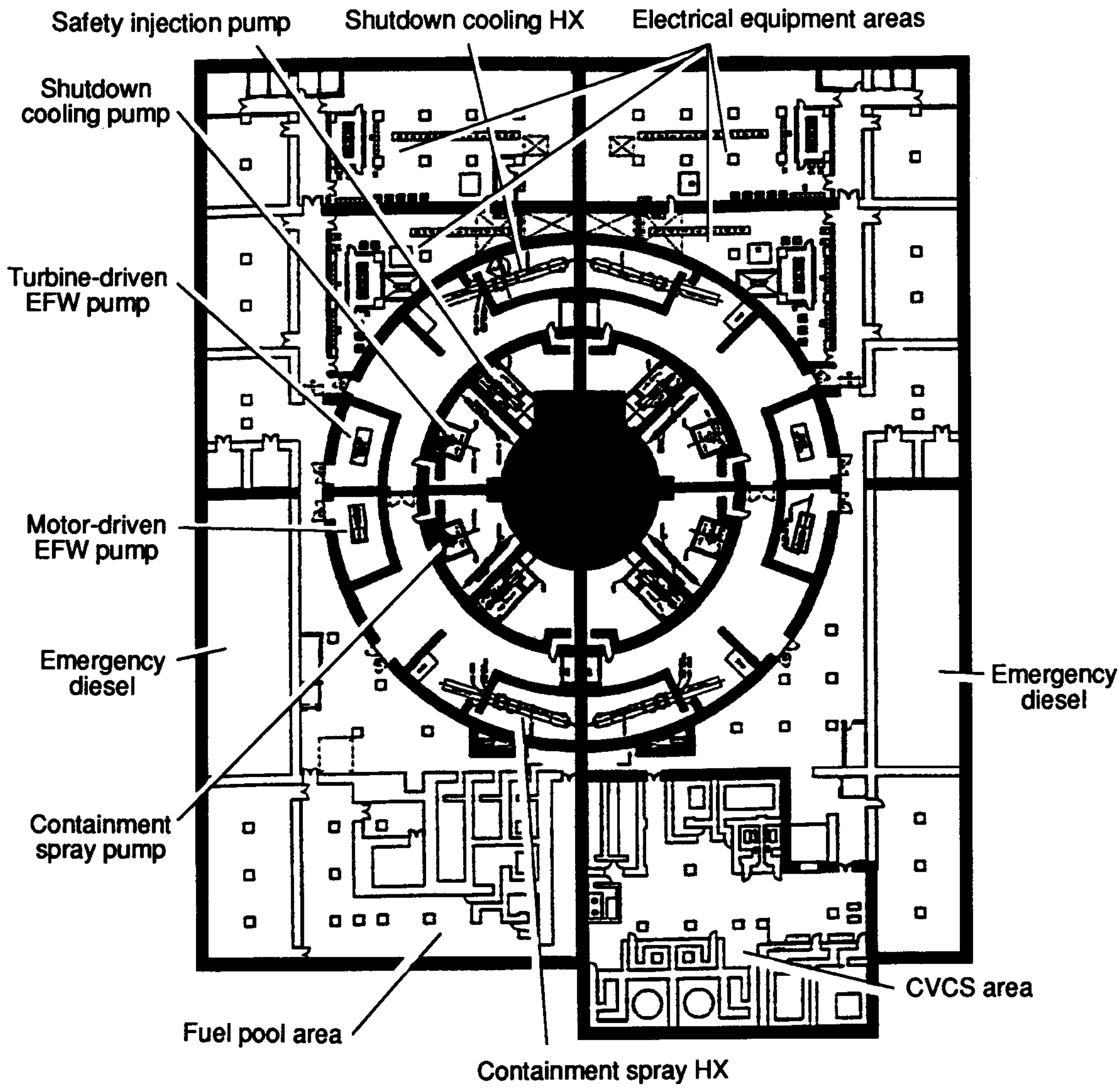

Fig. 4 General arrangement of containment and nuclear annex. EFW, emergency feedwater; HX, heat exchanger; CVCS, chemical and volume control system.

to power with steam line break (Fig. 11). These results demonstrate that fuel damage and offsite exposure are virtually eliminated for the System 80+ Standard Plant Design.

DESIGN FOR SEVERE ACCIDENTS

A severe accident is one that involves appreciable core damage. In concert with attention to public health and safety, the focus of severe accident design is investment protection. Combined with regard for the human operator, this is perhaps the most important lesson of the Three Mile Island Nuclear Station accident. System 80+ is a more resilient plant designed not only to prevent core damage but also to moderate the severity of such an accident should it occur. This is the function of containment and the systems that support it.[38] A 61-m-diameter and 96 000 m^3 of free volume allow cost-effective innovation to directly address severe accident concerns; selected features include a reactor cavity that ensures coolability and retention of molten core debris, a passive cavity flooding system, and hydrogen ignitors that operate independent of site power. A safety depressurization system (SDS) is added to prevent containment failure caused by direct containment heating from high-pressure core-melt ejection.

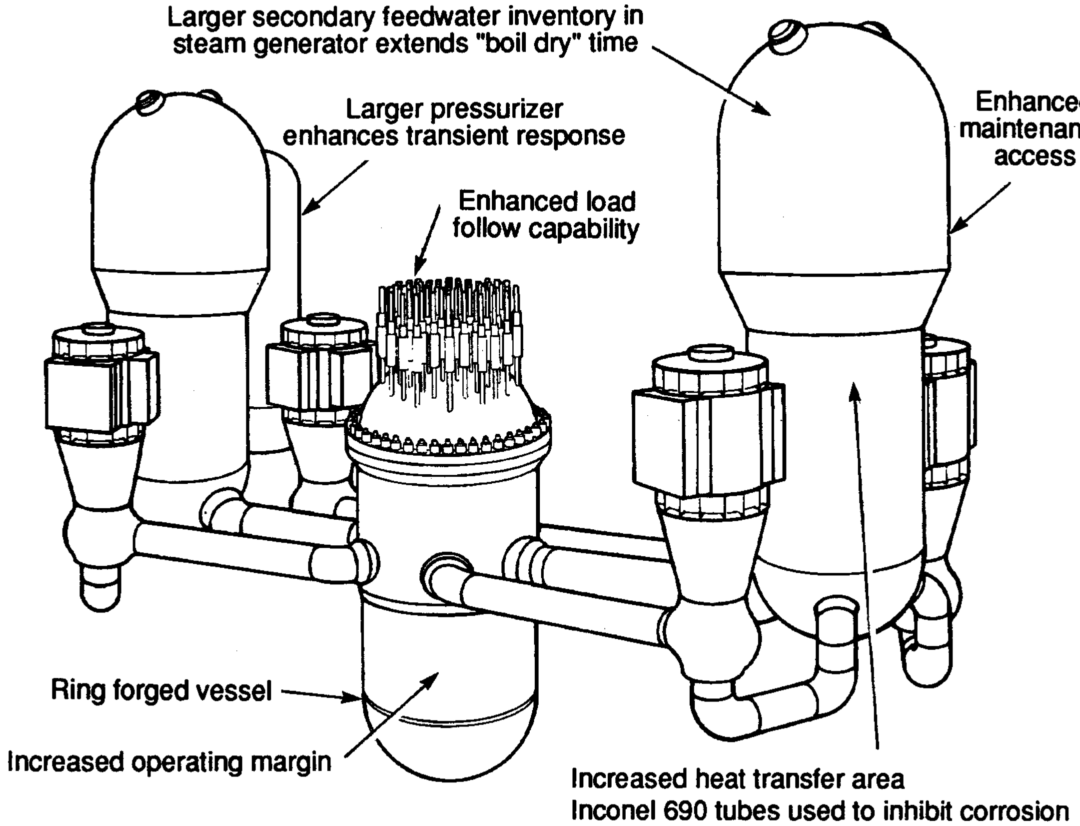

Fig. 5 Principal reactor coolant system improvements.

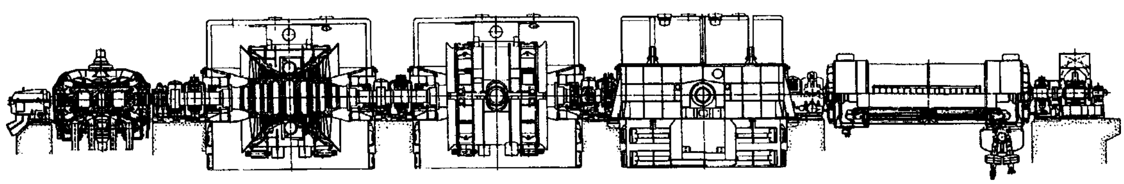

Fig. 6 Typical 1300-MW steam turbine for ABB nuclear power plants. HP, high pressure; LP, low pressure.

Reactor Cavity and Flooding System

The reactor cavity configuration is designed to prevent debris transport and to provide coolability; it incorporates an exit area greater than the area around the vessel, a collection volume twice the core volume, and a floor area greater than 0.02 m^2/MW(t). The flooding system incorporates passive gravity flow from the IRWST to the cavity via a holdup volume.

Hydrogen Control

The large System 80+ containment is designed to prevent hydrogen buildup by natural circulation and can passively accommodate a metal–water reaction of up to 75% of the core metal without exceeding a hydrogen concentration of 13% by volume.[39] Ignitors are provided to meet current NRC requirements to accommodate a 100% reaction and maintain hydrogen below 10%.

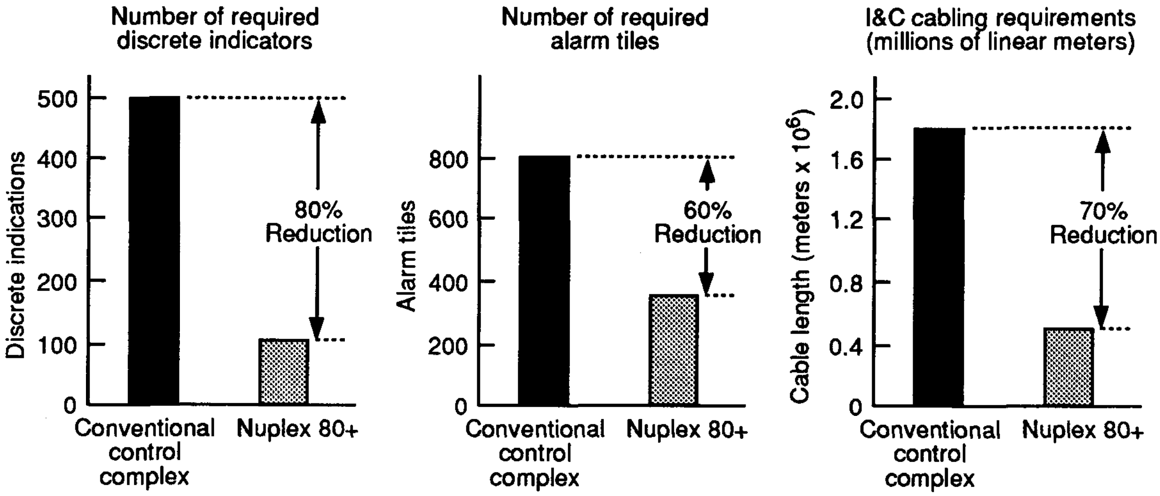

Fig. 7 Nuplex 80+ benefits. I&C, instrumentation and control.

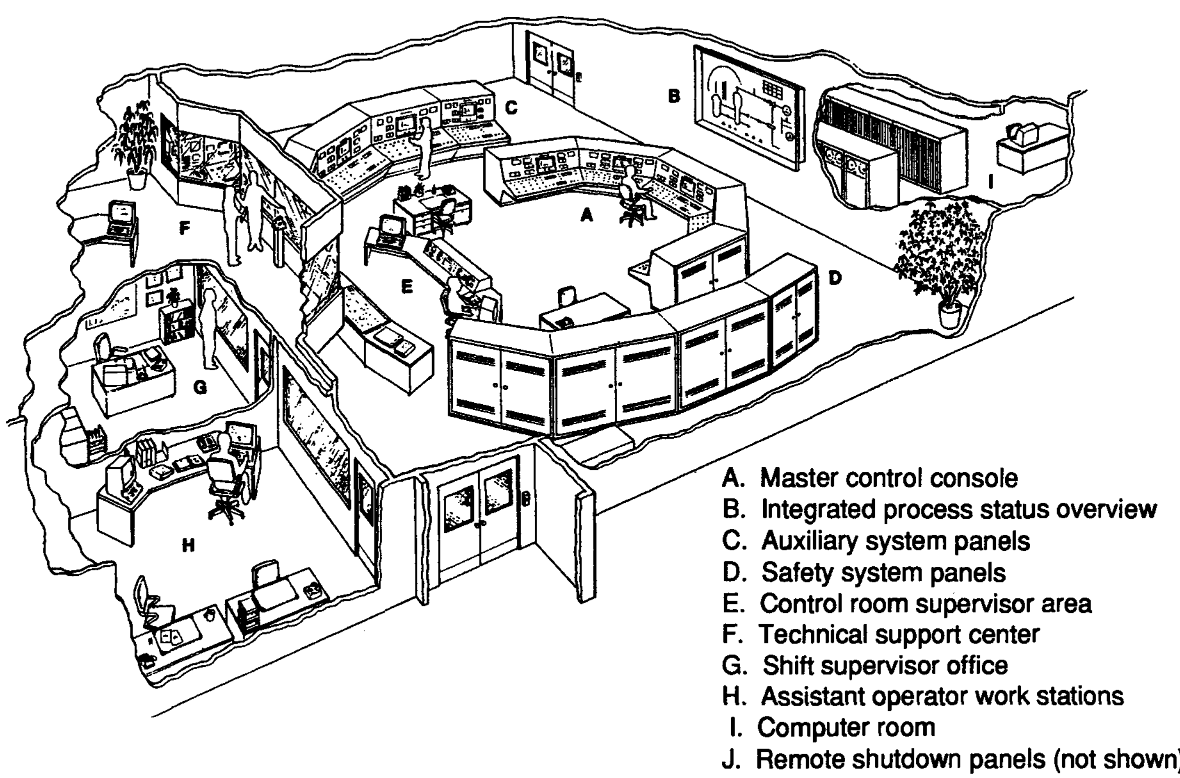

Fig. 8 Nuplex 80+ advanced control complex.

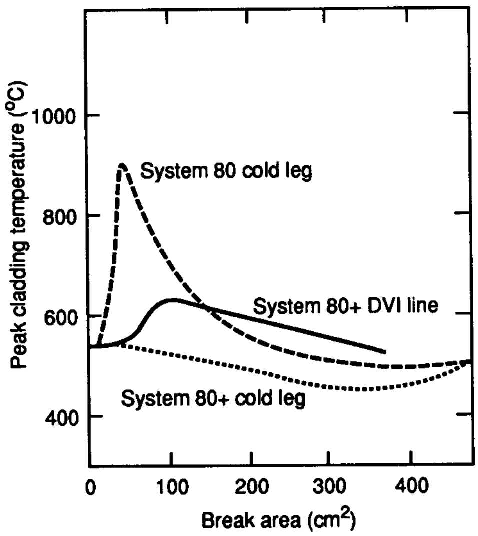

Fig. 9 **Small-break loss-of-coolant accident analysis, peak cladding temperatures. DVI, direct vessel injection.**

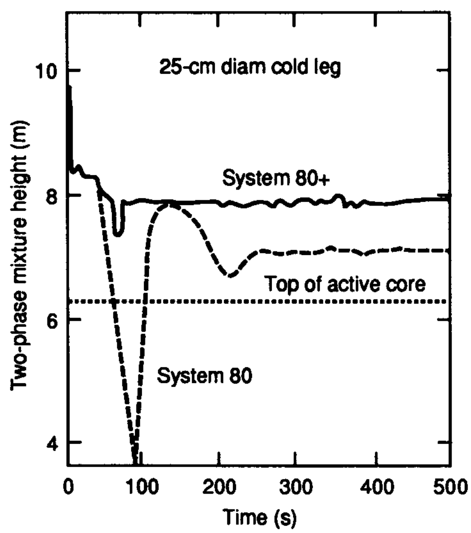

Fig. 10 **Small-break loss-of-coolant accident.**

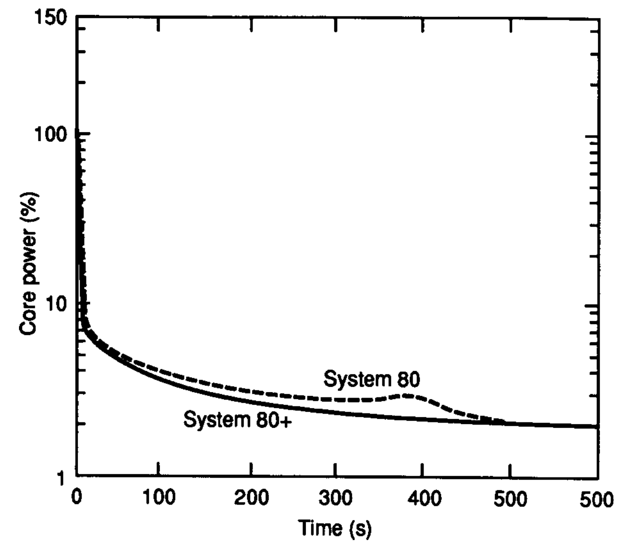

Fig. 11 **Main steam line break accident.**

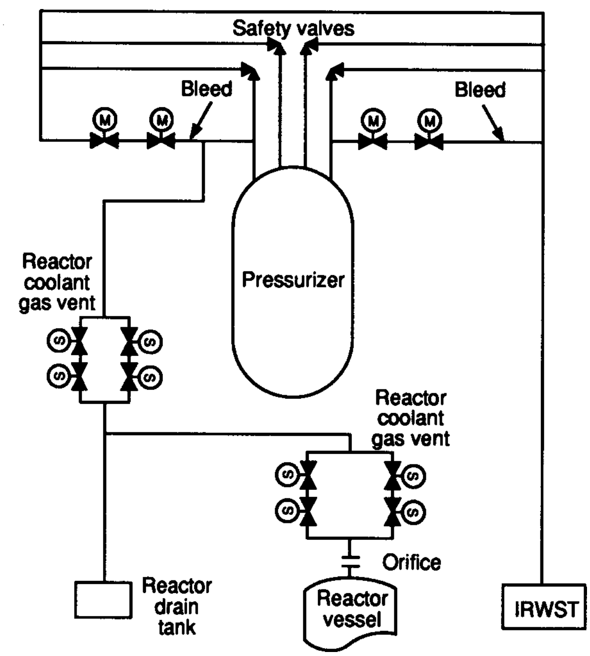

Fig. 12 **Safety depressurization system. S, solenoid.**

Safety Depressurization

A dedicated SDS provides an alternative decay heat removal path through primary feed and bleed (Fig. 12). This offers a means to rapidly decrease pressure and thereby keep the core covered even when all feedwater is cut off. System pressure can be reduced from 170 to 30 bar in less than 2 hours. Other benefits include normal decay heat removal and safety-grade depressurization during design-basis events. The SDS provides a safety-grade backup to pressurizer spray for cooldown and steam generator tube rupture.

Results

Probabilistic risk assessment (PRA) focuses on core damage prevention not only to preserve health and safety but also to consider the integrity of plant investment in

Table 1 Mean Core Damage Frequency Per Reactor Year[a]

Initiating event	System 80	System 80+
Small LOCA	9.4×10^{-6}	4.4×10^{-8}
Medium LOCA	3.6×10^{-6}	9.1×10^{-8}
Large LOCA	1.6×10^{-6}	5.0×10^{-8}
Steam line break	9.0×10^{-7}	2.0×10^{-10}
Transients	1.2×10^{-5}	3.3×10^{-8}
LOOP/SBO	3.8×10^{-5}	1.0×10^{-7}
SGTR	1.0×10^{-5}	8.0×10^{-8}
Vessel failure	1.0×10^{-7}	1.0×10^{-7}
ATWS	4.8×10^{-6}	1.7×10^{-7}
Interfacing LOCA	4.5×10^{-9}	3.0×10^{-9}
Internal events	8.1×10^{-5}	6.7×10^{-7}
External events	3.0×10^{-5}	1.2×10^{-6}
Total	1.1×10^{-4}	1.9×10^{-6}

[a]Acronyms used:

ATWS	Anticipated transient without scram
LOCA	Loss-of-coolant accident
LOOP	Loss of offsite power
SBO	Station blackout
SGTR	Steam generator tube rupture

ensuring economic health for the electric utility and the community it serves.[40] Final PRA results for System 80+ show two orders of magnitude reduction in severe accident risk compared with System 80 (Table 1).[36] This low risk of core melt compares very favorably with that of other advanced designs (i.e., equal to or smaller than), including the passive ALWR. The design is more balanced with respect to individual contributors to severe accident risk (Fig. 13). Half the result is due to the addition of an SDS, which provides a diverse means of decay heat removal. With feed-and-bleed capability, depressurization valves rapidly reduce pressure to below the shutoff head of the safety injection pumps in the event of a total loss of feedwater flow, and thus 1.1 m of water is maintained over the core at all times. With advances in the electrical distribution system (EDS), including addition of an alternate a-c power source, loss of offsite power/station blackout (LOOP/SBO) is virtually eliminated. Combined with safety injection and emergency feed improvements, 90% of the risk reduction is secured (Fig. 14).

CONCLUSIONS

System 80+ represents major advances in the technology of light-water-reactor safety. Accidents are far less

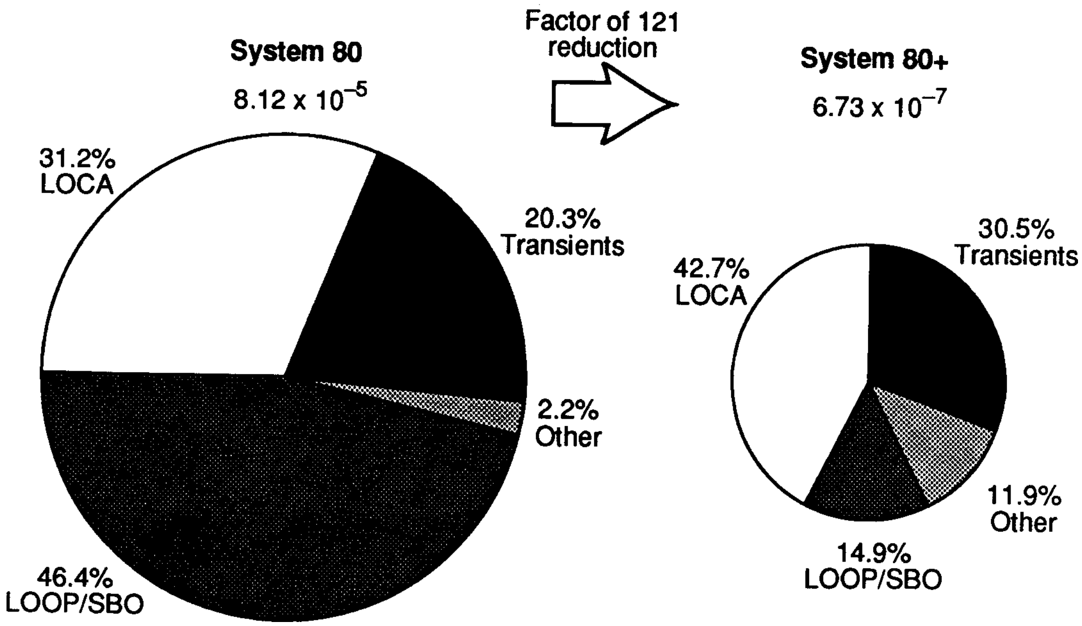

Fig. 13 Dominant contributors to severe accident risk (core damage frequency, internal events). LOCA, loss-of-coolant accident; LOOP, loss of offsite power; SBO, station blackout.

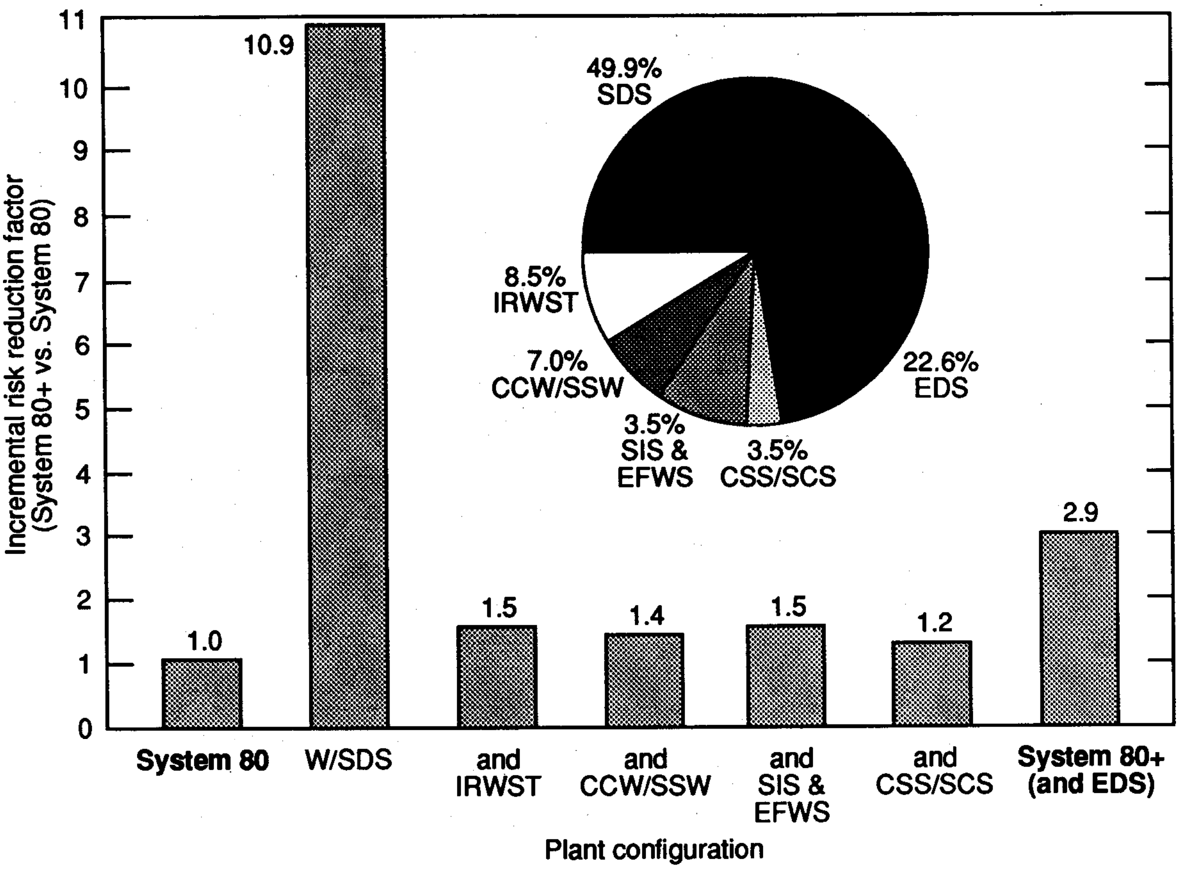

Fig. 14 Impact of System 80+ design features on severe accident risk (core damage frequency, internal events). W/SDS, with safety depressurization system; IRWST, in-containment refueling water storage tank; CCW/SSW, component cooling water/ station service water; SIS, safety injection system; EFWS, emergency feedwater system; CCS/SCS, containment spray system/ shutdown cooling system; EDS, elecfrical distribution system.

probable as the result of a balanced and integrated approach to nuclear plant design that considers both prevention and mitigation in the appropriate context. Responses to potential accidents are slowed down, easier to manage, and result in consequences that are far more benign.

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Environmental Effects

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The MATS Experiments—Mesoscale Atmospheric Transport Studies at the Savannah River Site

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Abstract: *An overview of the Mesoscale Atmospheric Transport Studies (MATS) program is presented. MATS was an experimental program to create a data base for short-term atmospheric releases to study mesoscale atmospheric dispersion over gently rolling terrain at the Savannah River Site (SRS). Thirty-one experiments were performed over a 3-year period under daytime convective conditions using 15-minute releases of sulfur hexafluoride (SF_6) tracer. Dispersion was typically measured along an arc of stationary samplers approximately 30 km downwind. Four nighttime experiments were performed with SF_6 releases, and sampling was carried on for a few hours with a mobile sampling laboratory. The mobile laboratory repeatedly traversed the plume while recording tracer concentration, time, and location.*

Supporting meteorological data for the MATS experiments included wind and temperature measurements from an integrated meteorological tower network and computer system. The tower network consists of eight instrumented, 61-m towers located near each of SRS's major facilities and a 304-m television tower located about 16 km WNW from the SRS administrative area. Low-level soundings obtained from on-site launches of mini-rawinsondes and Airsondes provided additional data on the vertical structure of the boundary layer for 11 of the experiments. Soundings were also obtained for Athens, Ga., and Charleston, S.C., from the National Climatic Data Center. The MATS data set is available for use by diffusion researchers and modelers.

Two experiments were examined in detail and compared with predictions from a simple Gaussian model to help demonstrate the usefulness of the data. These cases show that the downwind locations of tracer material were predicted adequately if uniform winds at release level were assumed but that the magnitude of the concentration was overpredicted by factors of 2.5 and 10.

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The Westinghouse Savannah River Company (WSRC) maintains an emergency response system for the Department of Energy's Savannah River Site (SRS). The system is operated by the Savannah River Laboratory (SRL) and includes dispersion codes to predict the path and concentration of unplanned airborne releases of radionuclides or chemical pollutants. A general description of the atmospheric support activities at SRS was given by Addis and Hunter.[1] Descriptions of the emergency response system were presented at the American Nuclear Society's Topical Meeting on Radiological Accidents in 1986 (Refs. 2 to 5).

There has been an ongoing effort to evaluate SRL's dispersion codes with real data. Dispersion experiments performed over the past 15 years at SRS have increased the understanding of atmospheric transport and dispersion and improved models for a variety of transport distance scales. For example, the Savannah River Experiment (SRE) (Ref. 6) provided a data base to test mesoscale dispersion models with 10-hour and weekly time-averaged samples. The SRE studied releases of ^{85}Kr from SRS reactor stacks that were monitored by periodically collecting air samples at 13 sites in the distance range 25 to 150 km over a 2.5-year period. The Atlantic Coast Unique Regional Atmospheric Tracer Experiment (ACURATE) provided data from these stacks for regional scale dispersion. ACURATE was done in conjunction with the U.S. Department of Commerce, National Oceanic and Atmospheric Administration, Environmental Research Laboratories.[7]

Evaluation of SRS's mesoscale dispersion models requires data from both short- and long-term releases.

Although a wide variety of atmospheric dispersion experiments have been performed and analyzed since the 1950s with different tracers, sampling times, terrain, and release heights,[8] the majority of these experiments have not addressed the problem of relatively short-term (less than 1 hour) releases. The Mesoscale Atmospheric Transport Studies (MATS) program was conducted by SRL between January 1983 and August 1986 to provide a data base to study mesoscale dispersion for short-term releases at about 30 km downwind. The objectives of the MATS program were to evaluate the performance of atmospheric dispersion models, to increase understanding of atmospheric dispersion downwind of SRS, and to develop and test emergency response capabilities, such as logistics, communications, forecasting skills, and sampling strategies.

Sulfur hexafluoride (SF_6) was used as the tracer gas in each MATS experiment. SF_6 was selected because of its safety, relatively low cost, inertness, low losses caused by solubility—a Henry's Law coefficient of 3.13×10^6 (mm Hg) (liters of water)/mole of gas[9,10]—and ease of detection (with electron-capture gas chromatography). The SF_6 was measured with a flowmeter and/or weighed with a load cell before being released into the effluent stream at the base of a 61-m stack for a 15-minute period.

In 32 experiments a line of stationary whole-air samplers was set out along roadways downwind from the release point near the center of the SRS. These samplers collected ambient air samples over a period of a few hours. In two experiments a mobile sampling vehicle was used in conjunction with the stationary samplers. In five experiments the mobile sampling vehicle was used without the stationary samplers. In total, 37 MATS experiments were performed.

SITE, MEASUREMENTS, AND EXPERIMENTAL DESIGN

The SRS occupies an area of 800 km^2 along the Savannah River in southwestern South Carolina, 40 km southeast of Augusta, Ga. (Fig. 1). Three reactors, two chemical reprocessing plants, a tritium processing and handling facility, a fuel fabrication facility, and a waste vitrification facility are all located on site. Major areas are designated as A area (administrative), F area (separations), etc., in Fig. 1. The terrain within and surrounding the SRS consists of gently rolling hills covered with pine forests, open fields, streams, and small lakes. Clearings within the SRS contain man-made structures, such as buildings, roadways, and parking lots, but occupy less than 5% of the SRS's total area.

All MATS experiments began with a release of SF_6 from a 61-m stack in H area (Fig. 2). H area was selected for the release point because it is situated near the center of the SRS and is encircled by a network of highways 26 to 40 km downwind. The highway system allowed easy deployment of fixed SF_6 samplers and operation of a mobile field vehicle that continuously sampled the air for tracer gas.

The vertical exhaust velocity at the top of the 2.4-m-diameter H area stack was ~8.6 m s^{-1}, which normally resulted in plume momentum rise of 5 to 60 m for the wind speed range 1 to 10 m s^{-1} using Briggs' formulas.[11] Most sampling during the MATS experiments was performed during the daytime at ground level about 30 km downwind of the source. At this distance downwind the effective stack height (physical stack height plus plume rise) usually has a relatively small effect on the ground-level concentration of a passive tracer.[12]

Four main components of SRL's meteorological facilities were used during these experiments: (1) the tower network for taking wind and temperature measurements; (2) the Weather Information and Display (WIND) system for meteorological data acquisition and running dispersion models; (3) the array of stationary whole-air samplers for intercepting the SF_6 plume; and (4) a mobile field laboratory, the Tracking Radioactive Atmospheric Contaminants (TRAC) vehicle, for real-time monitoring and analysis of SF_6 concentrations. Each of these components is briefly described.

Tower Network

The tower network consists of eight 61-m meteorological towers located at areas A, C, D, F, H, K, L, and P, and a 304-m instrumented television-transmitting tower (WJBF-TV). (See Fig. 2 for the locations of the towers.) During the MATS program each area tower was equipped with a fast-response bidirectional wind vane and propeller anemometer (MRI VectorVane) on a single boom. The VectorVanes were configured so that the propeller anemometer was mounted at the leading end of the bidirectional vane. The VectorVane allowed measurement of both horizontal and vertical wind direction angles as well as wind speed. The VectorVanes were mounted at an elevation of 61 m (except for the D area tower, which had the instruments mounted at both 18 and 61 m). The VectorVanes provided wind speed, azimuth, and elevation-angle observations that were recorded at 1.5-second intervals; 15-minute averages and standard deviations were then archived.

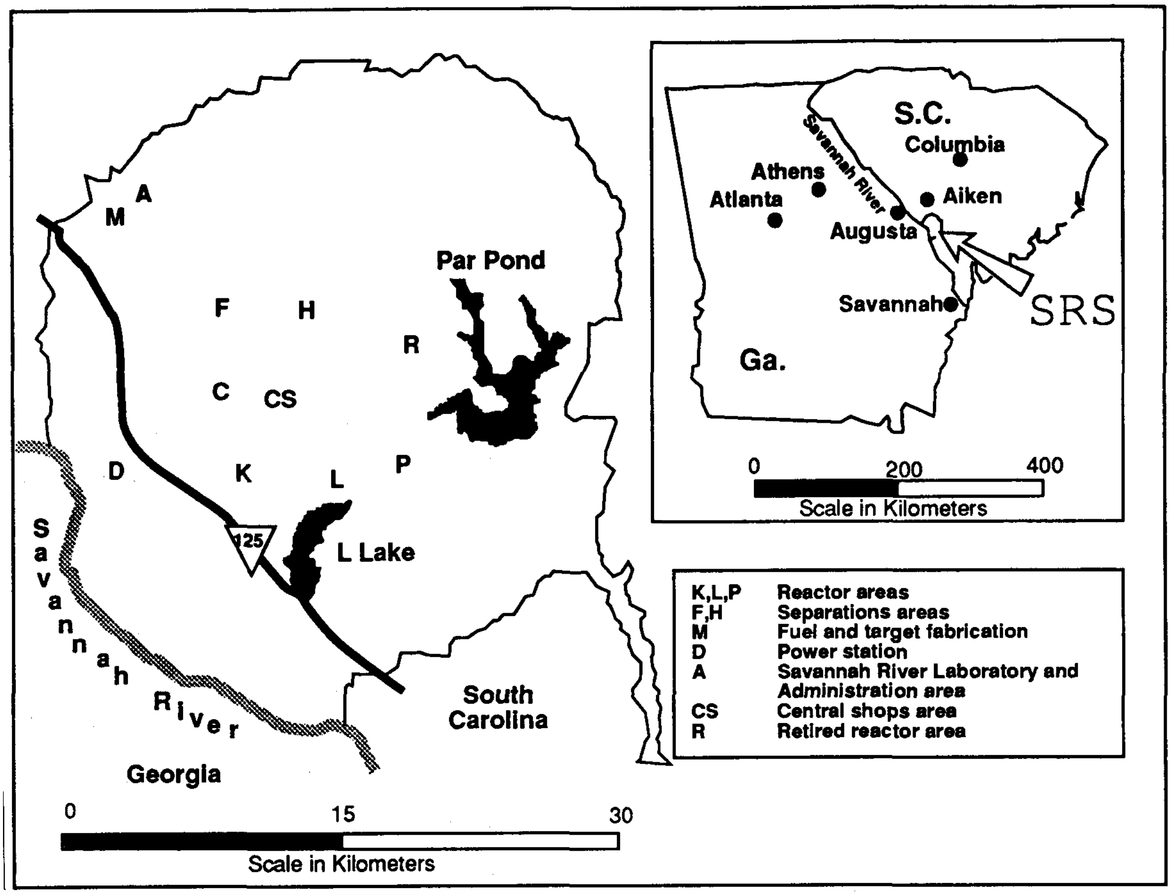

Fig. 1 Savannah River Site and locations of several of the major facilities at the site.

The WJBF television tower has Climet cup anemometers and bivanes at heights of 18, 36, 91, 137, 182, 243, and 304 m. The Climet cup anemometer is a three-cup anemometer that spins horizontally on a vertical shaft and thus causes a slotted disk to interrupt a beam of light impinging on a photocell. The rate of rotation is then determined by electronically counting voltage pulses. These anemometers have a starting speed of 0.26 m s^{-1} and are able to measure winds up to 40 m s^{-1}. The television tower also has aspirated platinum resistance thermometers at the same height as the wind instruments. Wind and temperature data were measured every 1.5 seconds and block-averaged over three readings; 15-minute averages and standard deviations were then archived.

WIND System

The WIND system is a linked network of computers that acquires field data for use in aqueous and atmospheric dispersion model codes. The WIND system also monitors stack radiological source-term sensors, radionuclide detection sensors, and various environmental sensors. The WIND system's central computer facility at SRL processes meteorological data that the WIND system collects. The central computers perform quality assurance checks on the data and archive 15-minute means and standard deviations of tower wind speed, direction, gusts, turbulence intensities, and temperatures. The archived meteorological data are used in emergency response dispersion models and for other SRS-related activities, such as site forecasting and climatological studies.

The WIND system also retrieves regional meteorological data from the National Weather Service's Automated Field Operation Services (AFOS). These data include synoptic analyses, National Weather Service (NWS) Model Output Statistics (MOS) specifically computed for the SRS using WJBF television tower data, and upper-air soundings from nearby rawinsonde stations. The MOS predictions are 12-hour statistical forecasts of

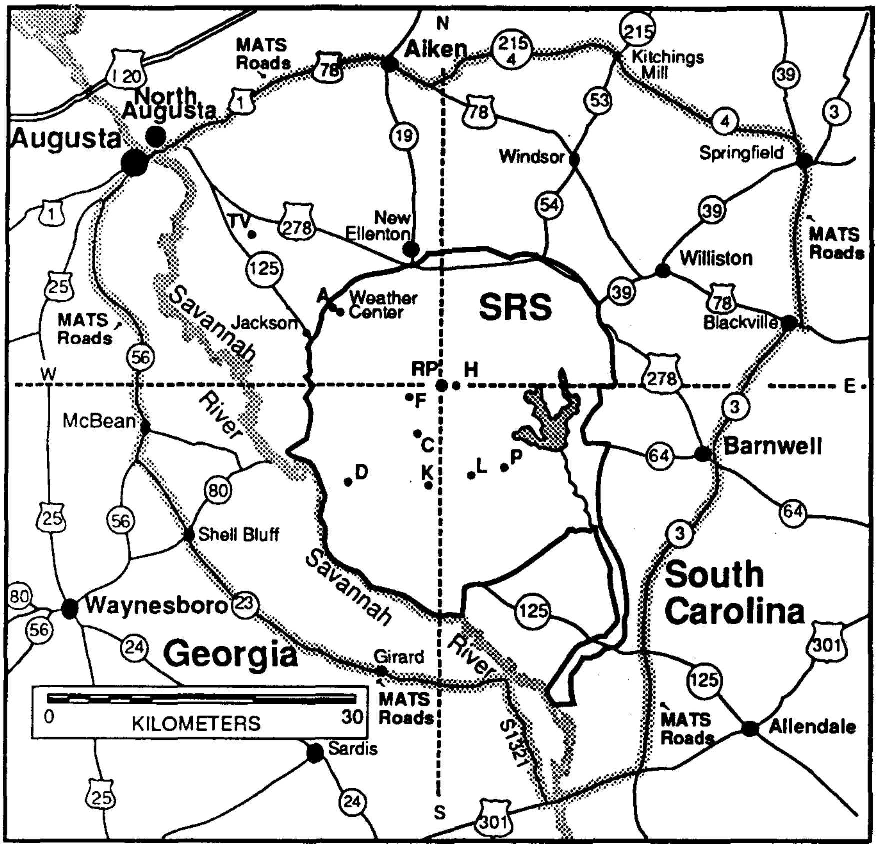

Fig. 2 Locations of the onsite meteorological towers, A,C,D,F,H,K,L, and P; the 304-m television tower (TV); and the roadways (shown hatched) where whole-air samplers were placed during the MATS experiments. The SF_6 release point (RP) was near the H area tower.

wind speed, direction, and turbulence intensities at three elevations for the boundary layer at SRS.

Three atmospheric dispersion models are included in the WIND system at present: a quick-executing straight-line Gaussian model, a fast-executing single-puff Gaussian trajectory model, and a multiple-puff Gaussian trajectory model with dose assessment capabilities. (Plans are proceeding to implement a fully three-dimensional prognostic model.)

Stationary Sampler Array

During the MATS daytime releases, two vans with 10 to 15 samplers were dispatched to the intersection point of the predicted tracer cloud trajectory and the encircling highway system, called the MATS arc, shown in Fig. 2. Once in position, two teams deployed a line of 15 to 30 sequential SF_6 whole-air samplers at regularly spaced intervals, typically 1.0 km apart. Vehicle odometer readings were used to mark off distances. After the 14th MATS experiment, a loran system was used to determine sampler locations more conveniently. (The loran system determines the position electronically by using triangulation from fixed radio transmission sites.)

In the first 10 MATS experiments the samplers consisted of a suitcase containing seven evacuated aluminum canisters connected to a common manifold and filled by electronically opening a valve to a pressure regulator. Additional electronic components caused successive sequencing from one canister to the next. The time needed to fill the seven canisters was determined from the estimated transit time of the tracer cloud. The sampling time for each canister was determined by equally dividing the transit time of the tracer cloud (allowing for a margin of error at the beginning and end of the experiment). Then the pressure regulator was adjusted to fill each canister at the proper rate. If all estimates were made skillfully and the winds did not vary unexpectedly, the first canister opened before the tracer cloud arrived and the last closed after it disappeared. The sampling time for each canister was typically 15 minutes.

Later MATS experiments included (Demaray) syringe-style whole-air samplers. These samplers contain 10 syringes attached to a common manifold. An electronic switching program causes tiny motors to fill each syringe sequentially with environmental whole-air samples. Because there are 10 syringes as opposed to 7 canisters, better time resolution of the tracer cloud is possible. The Demaray instruments could also be programmed more conveniently to set the sampling parameters.

At the end of each experiment the samplers were brought back to SRL for SF_6 gas chromatographic analysis. The air samples collected by the syringe-style instruments were easier to analyze chemically because no pumping of sample air was needed; each syringe could be injected into a gas chromatograph through a septum.

TRAC Vehicle

The vehicle is a 12-m-long mobile laboratory designed for plume tracking and real-time tracer sampling and analysis.[13] Communications were established between the Weather Center Analysis Laboratory (WCAL) located in A area and the TRAC vehicle during the MATS experiments. (Meteorologists directed each experiment from the WCAL since it contained the WIND system computer terminals, AFOS, and communications equipment.) The TRAC vehicle located the SF_6 tracer cloud with the help of predicted trajectories obtained from the Weather Center's WIND System and onboard gamma-radiation monitors that could detect plumes downwind from ^{85}Kr-emitting stacks in the reactor areas.

The TRAC's onboard laboratory contained an electron-capture continuous gas chromatograph (by AeroVironment, Inc.) to measure SF_6 in ambient air. The air was drawn in from a port on top of the vehicle. Known standards of SF_6 were checked before, during, and after each experiment. Onboard computers recorded signals from the continuous SF_6 analyzer and from the loran system. Results of the analysis were available as early as 1 hour after the return of the vehicle to the garage site.

Since the TRAC vehicle, the WCAL, and the sampling teams were in radio contact, field crews could receive updated information and adjust the programs of the fixed samplers or relocate the samplers if necessary. The TRAC vehicle also had the advantage that it could be used during the four nighttime MATS experiments when deployment of the fixed samplers was not practical because of safety concerns.

Other Measurements

In addition to the measurements described previously, standard rawinsonde data from the National Weather Service station at Athens, Ga. (shown in Fig. 1), were obtained and archived for most experiments. A limited number of rawinsondes and Airsondes were also released from A area near SRL. These soundings were taken to obtain higher resolution temperature, humidity, and wind profiles in the planetary boundary layer. The soundings are available for 12 of the last 18 MATS experiments.

THE MATS DATA BASE

General summaries of the MATS experiments are given in Tables 1 and 2. Roughly half the experiments were conducted in 1983 and the other half in 1985 and 1986. Of the 37 planned experiments, 2 were canceled or not completed (#22 and #35) because of mechanical difficulties and 4 others failed to produce measurable amounts of SF_6 downwind (#2, #3, #11, and #26) owing to logistical problems in intercepting the SF_6 tracer cloud. All but 4 of the 37 experiments were conducted during the daylight hours with SF_6 releases made in the late morning. As indicated in column 3 of Table 1, a variety of sampling techniques were employed: stationary samplers were used alone in 30 experiments; the TRAC vehicle alone was used in 5 experiments; and both techniques were used concurrently in the remaining 2 experiments (normally there were not enough workers to employ both sampling methods simultaneously).

Columns 5 and 6 of Table 1 provide SF_6 release rate and duration for the MATS experiments. All releases were made from a 61-m stack at H area (see point labeled RP in Fig. 2). Each release lasted 15 minutes. Column 7 gives concentrations multiplied by wind speed and divided by source strength (χUQ^{-1}) for the maximum concentration of SF_6 detected by any downwind sampler; χ is the maximum observed concentration (kg m^{-3}), U is the mean hourly wind speed (m s^{-1}) at 61 m, and Q is the release rate of SF_6 (kg s^{-1}). The last five columns of Table 1 summarize wind conditions at H area for the first hour of sampling.

The maximum value of the standard deviation of the crosswind concentration distributions, σ_y (column 8), was calculated from observed crosswind distributions of SF_6. The calculation was made by computing the second moment of the concentration distribution about the center of

Table 1 Summary of Release, Sampler, and Meteorological Data for the MATS Experiments[a]

MATS experiment No.	Date	Samplers used, No. and type	Start of release/ start of sampling, UTC[b]	SF_6 release rate, 1000 kg s^{-1}	SF_6 release time, min	Maximum normal concentration ($\times 10^{-6}$), m^{-2}	σ_y width, km	Hourly wind summaries at H area (61 m)				
								AZIM, degrees	SPD, m s^{-1}	GST, m s^{-1}	SIGA, degrees	SIGE, degrees
1	01/02/83	16E	1615/1705	35.1	15	4.32	1.20	027	5.7	9.3	10.6	9.0
2	04/06/83	18E	1611/1830	33.9	15	0	X	217	2.8	5.9	27.5	15.6
3	04/22/83	17E	1717/2000	32.5	15	0	X	199	3.9	5.6	12.6	10.7
4	05/20/83	19E	1515/1615	19.7	15	4.41	1.36	208	4.8	7.7	11.5	9.4
5	06/01/83	18E	1630/1730	40.6	15	3.78	1.82	347	5.1	8.1	12.9	8.5
6	06/08/83	19E	1416/1620	40.6	15	1.16	2.01	037	3.6	6.2	16.2	23.0
7	07/07/83	19E	1435/1600	78.3	15	3.98	2.30	041	4.6	8.5	18.8	14.7
8	07/22/83	18E	1530/1620	63.3	15	3.25	2.35	254	4.2	6.9	17.4	11.5
9	07/29/83	17E	1545/1630	66.1	15	1.06	2.29	061	4.3	9.3	21.9	16.1
10	08/16/83	18E	1415/1700	39.4	15	0.24	3.57	051	3.7	7.7	26.8	16.5
11	09/23/83	19S	1400/1455	86.1	15	0	X	080	2.9	4.8	14.0	12.1
12	09/28/83	19S	1430/1515	88.9	15	2.89	1.42	028	5.7	10.4	13.3	10.8
13	10/07/83	19S	1430/1500	27.8	15	4.31	0.84	059	7.3	11.7	10.9	9.2
14	10/17/83	18S	1400/1500	110.6	15	1.10	1.73	085	2.8	4.6	14.1	9.8
15	10/21/83	19S	1345/1445	101.0	15	7.23	1.28	027	4.1	7.3	13.2	11.0
16	12/02/83	29S	1545/1615	65.6	15	1.07	1.40	189	2.5	4.2	17.0	12.1
17	12/09/83	30S	1515/1615	56.1	15	0.82	4.55	177	2.7	5.5	29.9	16.6
18	12/15/83	30S	1545/1700	91.6	15	4.50	1.18	270	4.5	6.8	13.3	8.6
19	12/20/83	27S	1600/1700	80.0	15	3.39	2.05	031	4.3	7.5	15.5	12.8
20	06/11/85	27S	1500/1600	92.0	15	0.70	1.70	234	3.6	5.6	16.6	10.5
21	06/26/85	28S	1315/1515	90.0	15	0.60	2.14	254	2.4	4.4	33.9	21.0
22	07/09/85		Experiment canceled									
23	08/06/85	28S	1500/1600	115.0	15	0.76	5.72	127	3.8	7.0	21.5	14.6
24	11/05/85	28S	1600/1700	67.0	15	2.91	1.95	277	6.2	10.3	16.5	12.1
25	11/25/85	28S	1515/1630	13.3	15	1.72	4.72	115	1.6	3.1	16.8	9.8
26	12/11/85	28S	1600/1632	69.4	15	0	X	207	5.0	8.8	17.3	10.7
27	12/18/85	27S; TR	1630/1630	133.0	15	1.51	2.34	291	2.2	4.0	28.4	16.1
28	01/14/86	28S; TR	1600/1715	33.3	15	2.30	5.49	234	6.6	11.5	15.3	10.5
29	01/22/86	28S	1600/1645	61.1	15	0.61	2.94	229	3.3	6.3	22.7	14.3
30	02/04/86	28S	1600/1715	103.0	15	1.23	1.71	238	5.5	9.2	17.6	11.7
31	02/19/86	28S	1600/1650	100.0	15	1.23	2.14	271	6.2	10.9	18.2	10.9
32	03/04/86	TR	0100/0100	133.0	15	1.67	0.70	333	6.6	11.5	11.7	9.3
33	03/25/86	TR	0813/0838	52.0	15	0.64	0.30	120	5.7	7.2	4.7	1.8
34	04/29/86	28S	1500/1540	108.0	15	1.18	1.55	348	3.4	7.2	29.3	14.5
35	06/06/86	TR	No TRAC data available									
36	07/02/86	TR	0545/0600	27.2	15	1.31	0.45	257	12.5	13.2	2.1	0.1
37	08/08/86	TR	0700/0700	24.2	15	1.67	1.92	202	5.8	7.0	6.2	2.1

[a]Abbreviations used:

- X Missing data
- E Evacuation-type fixed samplers
- S Syringe-type fixed samplers
- TR TRAC vehicle sampling
- SIGA Standard deviation of wind direction
- SIGE Standard deviation of wind elevation
- σ_y width Standard deviation of crosswind concentration distribution
- AZIM Mean wind direction
- SPD Mean wind speed
- GST Maximum gust speed

[b]UTC Differs from local standard time by 5 hours

mass of the distribution (taking into account the angle of the MATS roadway with respect to the arc from the release point).

Table 2 provides a summary of the data available for each MATS experiment. This table shows the availability of release data, sampler data, sampler coordinates, concentration measurements, NWS rawinsonde soundings, SRS meteorological data, TRAC data, and SRS supplementary Airsonde and/or rawinsonde data. File descriptions and additional information may be obtained in a user's guide for the MATS data.[14]

Research studies based on some of the earlier MATS experiments have already appeared in print. In particular, the first 19 MATS experiments were used for extensive model testing that was reported at the Department of Energy/American Meteorological Society's Air Pollution Model Evaluation Workshop.[15–17]

Modelers compared predictions from six dispersion models with the MATS data at the model evaluation workshop. The models tested included two Lagrangian particle-in-cell models, a second-order closure Gaussian puff model, and conventional Gaussian plume models. The modelers were asked to run their models with the meteorological measurements from the eight area towers, the instrumented television tower, and rawinsonde soundings from Athens, Ga., or Charleston, S.C. One of the modelers tested the effect of using vertically averaged winds through the boundary layer rather than those measured directly from the 61-m towers. A second modeler compared the results of directly derived concentration fields with those from rotated and translated concentration fields so as to yield the best agreement with the observations.

Model performance at the workshop was evaluated with the use of two methods. The first method was based on consideration of speed and direction errors of the movement of the tracer center of mass and simple statistical representations of the data, such as bias, standard deviation, and a figure of merit (FOM). The FOM was defined as the ratio of the volumes of the intersection to the union of measured and predicted concentration distributions (in time or space). Table 3 shows a summary of model performance with the use of the first method for a subset of the statistics. An attempt was made to distinguish between complex and simple models.[16] Three of the models were classified as complex, and the remaining three were classified as simple. Two of the complex models were in the upper half of the rankings, whereas the third complex model was at the bottom of the rankings.

The second method of evaluation of model performance was based on a statistical ranking scheme deter-

Table 2 Summary of the Release, Sampler, Tower, and Special Meteorological Data Available for Each MATS Experiment

MATS No.	Date	Data included[a]
MATS 1	01-27-83	A,B,C,D,E,F
MATS 2	04-06-83	A,B,C,D,E,F,1
MATS 3	04-22-83	A,B,C,D,E,F,1
MATS 4	05-20-83	A,B,C,D,E,F
MATS 5	06-01-83	A,B,C,D,E,F,2
MATS 6	06-08-83	A,B,C,D,E,F
MATS 7	07-07-83	A,B,C,D,E,F
MATS 8	07-22-83	A,B,C,D,E,F
MATS 9	07-29-83	A,B,C,D,E,F
MATS 10	08-16-83	A,B,C,D,E,F
MATS 11	09-23-83	A,B,C,D,E,F,1
MATS 12	09-28-83	A,B,C,D,E,F
MATS 13	10-07-83	A,B,C,D,E,F
MATS 14	10-17-83	A,B,C,D,E,F
MATS 15	10-21-83	A,B,C,D,E,F
MATS 16	12-02-83	A,B,C,D,E,F
MATS 17	12-09-83	A,B,C,D,E,F
MATS 18	12-15-83	A,B,C,D,E,F
MATS 19	12-20-83	A,B,C,D,E,F
MATS 20	16-11-25	A,B,C,D,E,F,J
MATS 21	06-26-85	A,B,C,D,E,F,J,
MATS 22	07-09-85	3
MATS 23	08-06-86	A,B,C,D,E,F
MATS 24	11-05-85	A,B,C,D,E,F,I
MATS 25	11-25-85	A,B,C,D,E,F
MATS 26	12-11-85	A,B,C,D,E,F,1
MATS 27	12-18-85	A,B,C,D,E,F,G,4
MATS 28	01-14-86	A,B,C,D,E,F,G,J
MATS 29	01-22-86	A,B,C,D,E,F,J
MATS 30	02-04-86	A,B,C,D,E,F,J,5
MATS 31	02-19-86	A,B,C,D,E,F,J
MATS 32	03-04-86	A,E,F,G,J
MATS 33	03-25-86	A,E,F,G,J
MATS 34	04-29-86	A,B,C,D,E,F,J
MATS 35	06-06-86	A,E,F,I
MATS 36	07-02-86	A,E,F,H,6
MATS 37	08-08-86	A,E,F,G,I,J

[a]A Release data
B Sample data
C Sampler coordinates
D Sampler concentration measurements
E NWS rawinsonde soundings (Athens, Ga., sounding)
F SRS meteorological data
G TRAC traverse data
H TRAC special data
I SRS Airsonde data
J SRS rawinsonde date

1 No SF_6 detected at the tracer cloud's forecasted intersection with the MATS arc.
2 No tower data from 13:00 UTC to end of experiment
3 Experiment had to be aborted
4 Only one fixed sampler detected SF_6
5 SRS rawinsonde lost contact after first data transmission
6 No locations available for SF_6 measurements

Table 3 Summary of Simple Model Evaluation Statistics with the MATS Data[16]

Model[a]	Sp[b]	Dir[c]	Dist[d]	FOM[e]	Complexity[f]
SCIMP	−0.1	0.6	0.04	0.42	C
INPUFF	0.4	1.2	0.03	0.31	S
ADPIC	0.7	2.5	0.09	0.23	C
MSPUFII*	0.5	3.0	0.12	0.18	S
SCIMP*	−0.2	−3.9	0.14	0.16	C
MESOI	0.9	4.6	0.12	0.13	S
MSPUFII	−0.5	−1.1	0.17	0.05	S
ANLPUFF	1.1	6.6	0.13	0.11	C

[a]SCIMP: An integrated second-order closure Gaussian model. Concentration fields were rotated and translated to yield the best agreement with observations (Aeronautical Research Associates of Princeton). SCIMP*: Same as SCIMP except there was no rotation and translation of the concentration fields. INPUFF: An integrated puff model of traditional design but with updated treatments of turbulent growth and momentum rise (U.S. Environmental Protection Agency). ADPIC: A particle-in-cell model with excellent spatial and temporal resolution of wind and turbulence fields (Lawrence Livermore National Laboratory). MSPUFII: A conventional Gaussian plume model except that wind vectors could be input for two layers (National Weather Service and SRL winds vertically averaged through the boundary layer) (Environmental Research and Technology Co.). MSPUFII*: Same as MSPUFII except winds were only selected from SRL 60-m towers. MESOI: A standard Gaussian model except that vertical and horizontal dispersion coefficients were determined by separate algorithms (Pacific Northwest Laboratory). ANLPUFF: A six-particle puff model somewhat similar to ADPIC (Argonne National Laboratory).

[b]Sp: Average model speed bias (m s^{-1}) relative to the measured center of mass of the SF_6 distribution.

[c]Dir: Average model direction bias (degrees) relative to the measured center of mass of the SF_6 distribution.

[d]Dist: Average model distance bias (km) relative to the measured center of mass of the SF_6 distribution.

[e]FOM: Figure of merit, defined as the ratio of the volumes of the intersection and union of measured and predicted concentration distributions (in time or space).

[f]Complexity: A somewhat arbitrary designation (C, complex; S, simple) based on input wind fields, model resolution, model physics, and parameterizations of plume rise, dispersion coefficients, inversions, etc.

mined from seven to ten statistical parameters computed for each model. The parameters included the usual Pearson correlation coefficient, the Kolmogorov–Smirnov D statistic (for determining whether two distributions are similar), the smallest absolute difference between the actual regression line (of predicted vs. measured) and a slope of one, the mean-square error from the regression analysis, the average absolute bias, the variance of the differences between measured and predicted concentrations, the average total error, the difference of the medians, an F-test, and a t-test.[16] After each model was assigned a total ranking on the basis of the sum of the individual rankings from each statistic, a statistical procedure (the Kendall–Friedman rank test[16,18]) was applied to determine if statistically significant differences existed among the models.

A sample of results obtained with the Kendall–Friedman rank test is shown in Table 4. This procedure resulted in models being placed in groups (usually two to four) and being compared pairwise. Within each group no statistically significant differences in model performance could be detected (at the 95% confidence level). Models in a given group that do not overlap other groups are statistically different from all models outside the group. Overlapped models show insufficient evidence of statistical difference between any pairing between or among groups. As can be seen from Table 4, two complex models appear at the top of the highest group and do not overlap models below, but the third complex model shows no statistically significant difference between the lowest ranked model.

Table 4 Model Groups Resulting from an Application of the Kendall–Friedman Rank Test[16,21]

Group			Model[a]	Complexity
1			ADPIC	C
1			SCIMP	C
1	2		SCIMP*	C
1	2	3	INPUFF	S
1	2	3	ANLPUFF	C
	2	3	MSPUFII*	S
		3	MSPUFII	S
		3	MESOI	S

[a]The models were ranked on the basis of the first seven statistics cited in the text. The concentration patterns were integrated across the MATS arc for each sampling time segment. The models within each group (1,2, and 3) showed no statistically significant differences (at the 95% level). By statistically significant it is meant that within a group there are no statistically significant differences in the rankings between models as they are compared pairwise with the remaining models in that group. Two or more groups may overlap, which means that the overlapped models do not differ from other models of those groups. However, models of any group that do not overlap any other group (e.g., ADPIC and SCIMP) are statistically different from models outside their group (MSPUFII*, MSPUFII, and MOSOI). Models that overlap in all three groups (INPUFF and ANLPUFF) are not statistically different from any other model. (Complexity designation is the same as that found in Table 3).

The workshop showed that both methods of model evaluation yielded similar conclusions about model performance—complex models performed marginally better than simple models. The workshop also showed that better model performance was attained by averaging the measurements either spatially or temporally (as SRL had concluded during an earlier model evaluation workshop using ^{85}Kr data from the Savannah River Experiment[19]). It also became apparent that further guidance was needed from the scientific community on which statistical tests should be given the most weight in assessing model performance for specific purposes.

TWO CASE STUDIES

Two contrasting experiments, #24 (Nov. 5, 1985) and #37 (Aug. 8, 1986), are presented, and the results are compared with those from a simple Gaussian model to illustrate the types of analyses possible with the MATS data. Experiment #24 was conducted in the daytime under moderate westerly winds. SF_6 concentrations were measured along a downwind arc with the Demaray stationary samplers. Experiment #37 was conducted at night under moderate southerly flow during which the TRAC vehicle measured SF_6 concentration by repeatedly driving across the tracer cloud.

The Model

The model used here to compare with the MATS data is a slight variant of the fast-executing single-puff Gaussian trajectory model used by SRL to provide the basis for decisions in emergency situations where undesired effluent is emitted from a stack. The model, PUFF-PLUME,[20] has rainout, fallout, radioactive decay, and dose assessment features, but those capabilities are not relevant for the two case studies described here. The model becomes a simple reflecting Gaussian puff (or plume) model when there is no deposition or radioactive decay (as would be the case with a conservative, passive tracer such as SF_6).

The puff dispersion parameters used in the model are computed from expressions developed by Smith.[12] The model can be operated in either puff or plume mode and is based on the assumption that the terrain is flat between the release and sampling points. The model was modified slightly from its original form to give a puff trajectory as a function of downwind distance with 15-minute averaged winds measured at a single location. The winds were taken from the H area tower measurements and were assumed to be spatially uniform as a first approximation. The puff is assumed to have no buoyancy because the mixture of SF_6 and stack air is relatively dilute and the effluent has no significant temperature difference from the environmental air. Momentum rise of the puff or plume is allowed.

MATS #24—November 5, 1985

On the morning of Nov. 5, 1985, the weather over the southeastern states was dominated by a vigorous, occluded cyclone centered over western Maryland. The associated counterclockwise flow of air produced persistent westerly winds at the 304-m tower with little wind shear above the surface layer (Fig. 3). Two hours before the planned 15-minute release of SF_6 (scheduled for 1600 UTC), when meteorologists were developing a forecast for the intersection of the puff on the MATS sampling arc, the area towers reported a spatially averaged mean wind direction of 288° and mean wind speed of 6 m s^{-1} with gusts to 11 m s^{-1}. MOS predictions were for the wind's direction to remain constant but for the speed to increase to 9 m s^{-1} by midday. An Airsonde launched near the Weather Center at 1612 UTC showed the boundary layer to be about 800 m thick (Fig. 4).

Adopting the MOS guidance of a 9-m s^{-1} wind from compass direction 288°, we estimated that the center of the SF_6 cloud would reach the sampling arc 28 km downwind about 50 minutes after release with the cloud's leading edge arriving in about half that time. Two sampling teams were dispatched immediately to the section of highway directly downwind of the SF_6 release point (see Fig. 5). After arriving there, the teams set out 28 syringe-type whole-air samplers, 0.8 km apart, centered on the SF_6 tracer cloud's expected intersection point. The 10 syringes within each sampler were programmed to sample sequentially with a 12-minute period per syringe. The samplers were started at 1613 UTC—early enough to ensure that the entire SF_6 cloud would be detected as it crossed the sampling line.

The SF_6 was collected 2 hours along a line nearly 22 km long. Figure 6 shows a three-dimensional plot of the SF_6 concentration distribution. An overall impression of a Gaussian shape can be seen, but minor secondary maxima are present as well.

Comparison of Model with MATS #24

Analysis of the time-integrated SF_6 concentration collected by the samplers gave information on the position and time that the tracer cloud centerline intersected the sampling arc. Figure 5 shows the predicted cloud center

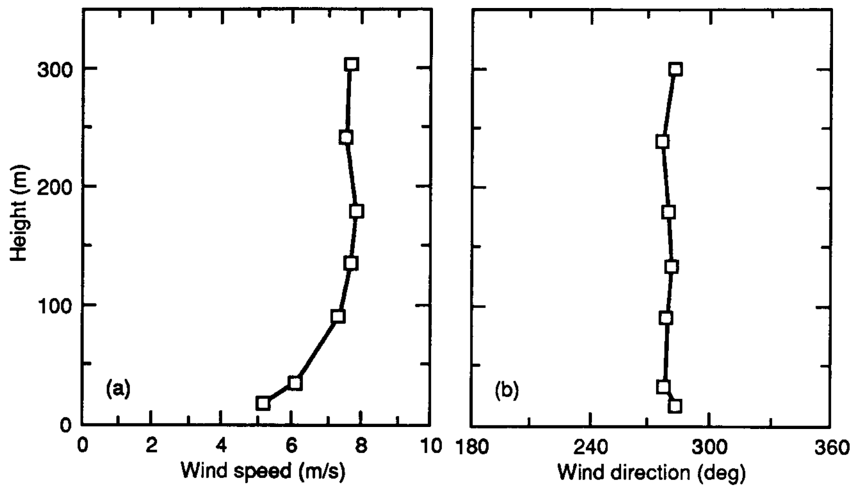

Fig. 3 MATS #24: The average hourly wind speed (a) and wind direction (b) from the 304-m television tower for Nov. 5, 1985, for 1600 to 1700 UTC (UTC differs from local standard time by 5 hours).

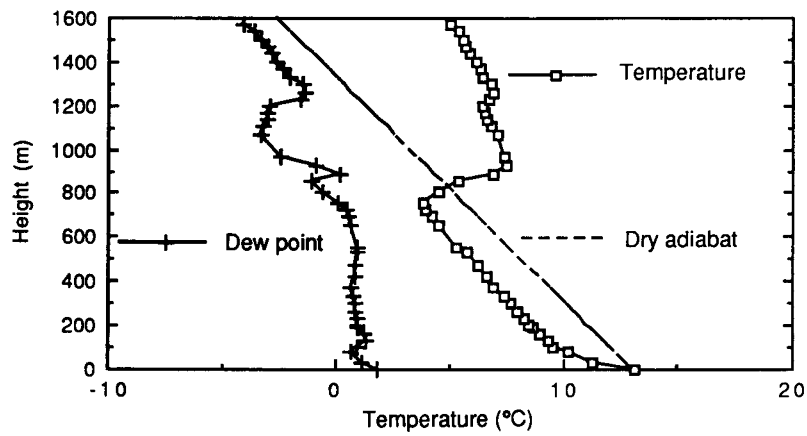

Fig. 4 MATS #24: Temperature–dewpoint sounding from an Airsonde release made near the Weather Center Analysis Laboratory on Nov. 5, 1985, beginning at 1612 UTC (UTC differs from local standard time by 5 hours).

crossing point on the MATS sampling arc, the puff trajectory based on H area winds, and the center of the measured time-integrated concentration distribution. The predicted and measured centerlines differ by approximately 7 km, but the measured centerline and the puff trajectory based on winds measured at H area are within 2 km. Note that a 10° error in the wind direction results in a 4.9-km error in the tracer cloud's intersection point with the sampling arc.

Figure 7 shows that the bulk of the (time-integrated) SF_6 was collected by samplers #15 to 26 (located north of the predicted crossing shown in Fig. 5). (The small amount of tracer collected in sampler #6 is thought to be an error since the amount detected was small and the gas chromatograph was measuring near the background concentration of SF_6.)

If, instead, crosswind-integrated SF_6 concentrations were to be analyzed syringe by syringe, a time history of the SF_6 cloud would be obtained. Thus Fig. 8 shows that the fifth syringe samples (integrated across the sampler arc) accumulated the maximum amount of SF_6. This implies that the tracer cloud's centerline intersected the sampling arc 60 to 75 minutes after the start of the release, or about 10 to 15 minutes later than predicted.

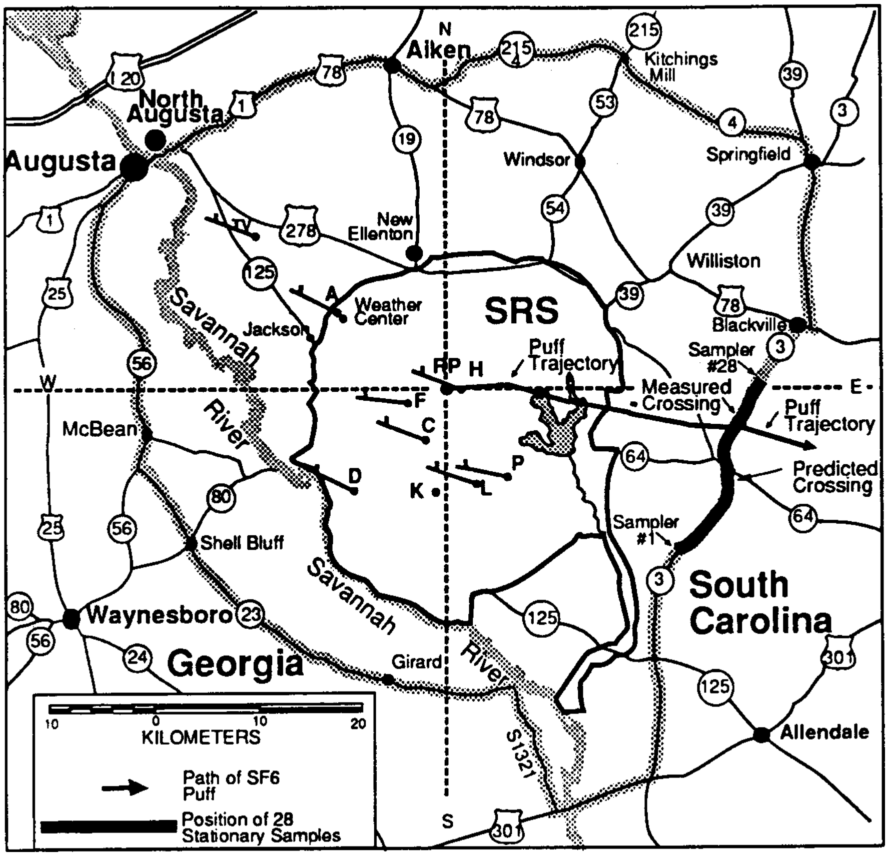

Fig. 5 MATS #24: Experimental conditions on Nov. 5, 1985, showing 15-min average winds (m s^{-1}) from the tower network at the time the forecast for sampling teams was issued (1400 to 1415 UTC). The predicted crossing of the tracer cloud with the MATS arc is shown. A puff trajectory based on 15-min averaged winds measured at H area beginning at 1600 UTC is also shown (UTC differs from local standard time by 5 hours).

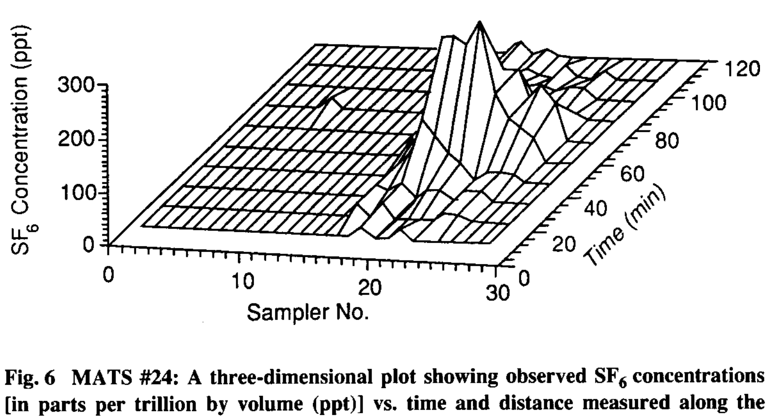

Fig. 6 MATS #24: A three-dimensional plot showing observed SF_6 concentrations [in parts per trillion by volume (ppt)] vs. time and distance measured along the sampling arc.

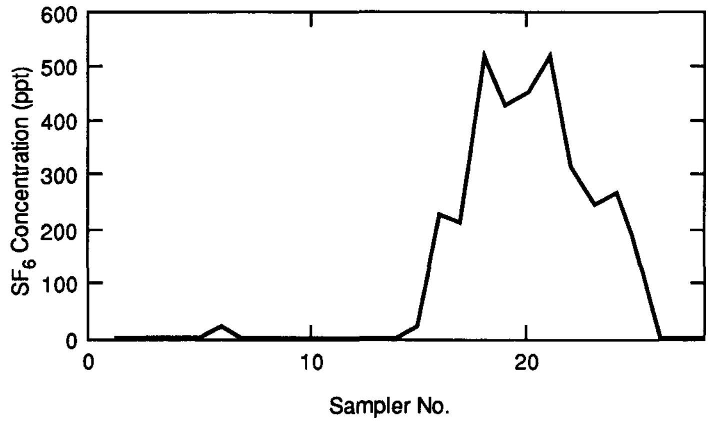

Fig. 7 MATS #24: Analysis of total time-integrated SF_6 concentration [in parts per trillion by volume (ppt)] collected by each of the 28 samplers.

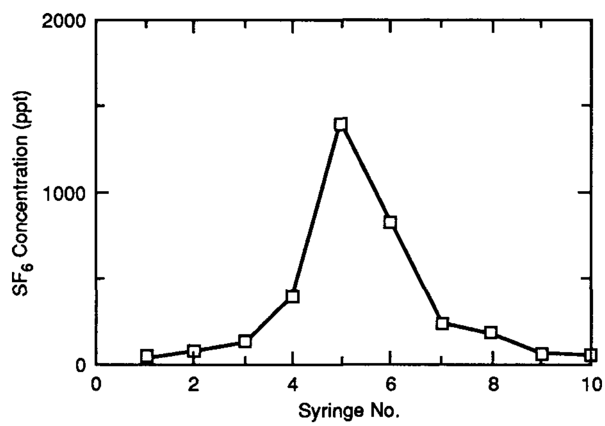

Fig. 8 MATS #24: Analysis of the crosswind-integrated SF_6 concentration [in parts per trillion by volume (ppt)] collected sequentially by the ten syringes in each sampler. The sampling time was 12 minutes per syringe.

The single-puff Gaussian model was run to determine the expected maximum concentration on the MATS arc. The longitudinal dispersion coefficient σ_x was given the same rate of growth as the lateral dispersion coefficient, as is commonly done.[12] The finite length of the cloud was taken into account by adjusting the longitudinal dispersion coefficient according to

$$\sigma_x = [\sigma_x^2 + (\text{finite length}/4.3)^2]^{1/2} \qquad (1)$$

following Pasquill.[12] The maximum concentration of SF_6 measured as the cloud passed was 253 parts per trillion by volume (ppt) as compared with 685 ppt predicted by the model.

Agreement between predictions and observations is within the expected range for meteorological dispersion calculations.[12] The minor discrepancies between predicted and measured centerline positions and arrival times are attributed to differences between our MOS prediction, which was used to help predict the tracer cloud centerline, and the actual wind flow. Wind speeds remained nearly constant at 6 m s^{-1} (instead of increasing to 9 m s^{-1}), and wind directions turned counterclockwise during the sampling period rather than remaining at 288° (see Fig. 9).

MATS #37—August 8, 1986

Surface synoptic maps on the evening of Aug. 7, 1986, and the morning of August 8 showed a wedge of high pressure extending westward from the Atlantic Ocean across Florida into the Gulf of Mexico. The accompanying clockwise circulation carried a flow of moist southerly air across South Carolina. A nighttime SF_6 release was planned for 0700 UTC with the use of the TRAC vehicle as a mobile sampling platform.

Winds at the 61-m area towers were southerly (190°) around 0300 UTC, turning more southwesterly (215°) as release time approached. Wind speeds averaged 5.5 m s^{-1} during the experimental period with relatively little deviation from hour to hour (see Fig. 10). Wind profiles obtained from the 304-m television tower (Fig. 11) showed significant shear in both speed and direction between 18 m and the top of the tower. MOS guidance indicated boundary layer winds would remain fairly

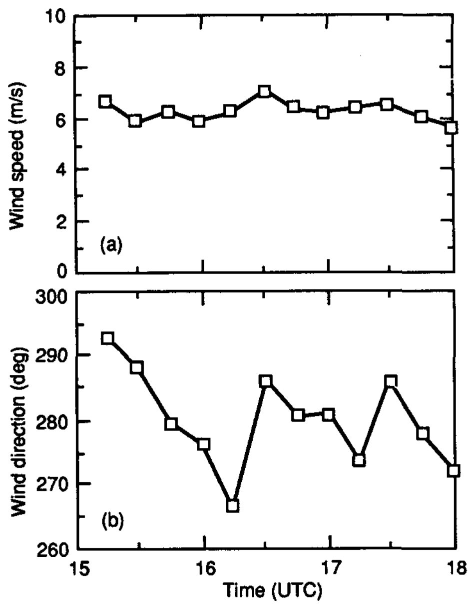

Fig. 9 MATS #24: Variation of wind speed (a) and direction (b) with time at H area (61 m) during the course of the experiment (Nov. 5, 1985) (UTC differs from local standard time by 5 hours).

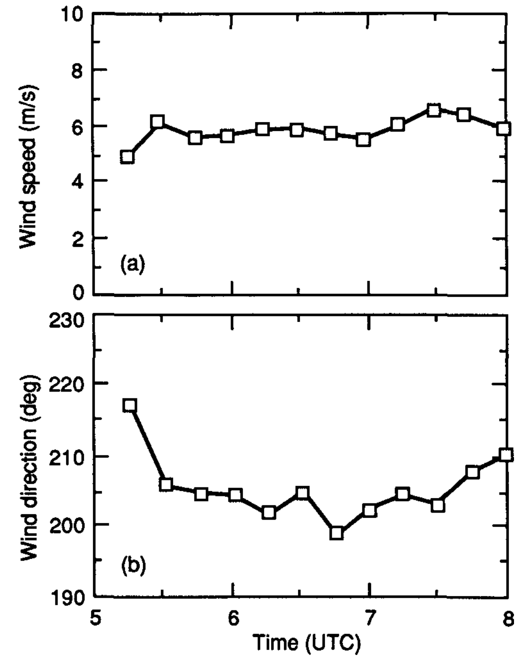

Fig 10 MATS #37: Variation of wind speed (a) and direction (b) with time at H area (61 m) during the course of the experiment (Aug. 8, 1986) (UTC differs from local standard time by 5 hours).

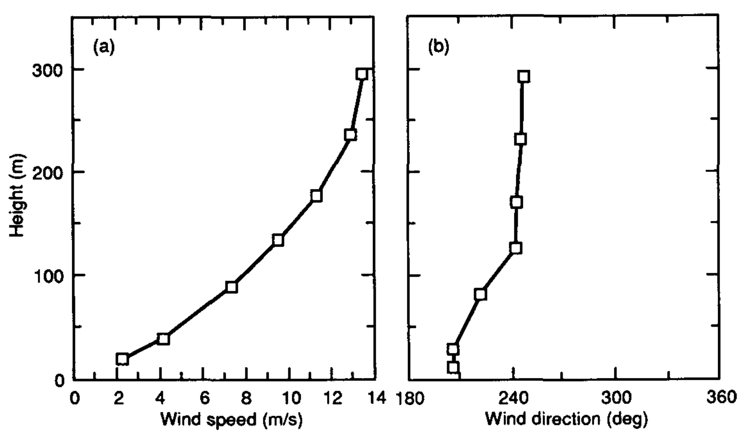

Fig 11 MATS #37: Average hourly wind speed (a) and direction (b) profiles from the 304-m television tower for Aug. 8, 1986, beginning at 0600 UTC (UTC differs from local standard time by 5 hours).

constant in speed and direction during the night before turning more westerly by 1200 UTC the next day. An Airsonde launched from SRS at 0530 UTC showed a nocturnal temperature inversion extending from the surface to an elevation of 214 m (Fig. 12).

Comparison of Model with MATS #37

The tracer cloud's leading edge was predicted to reach the MATS arc 33 km downwind in approximately 1.5 hours by using a 1-hour average of 5.5 m s^{-1} for the

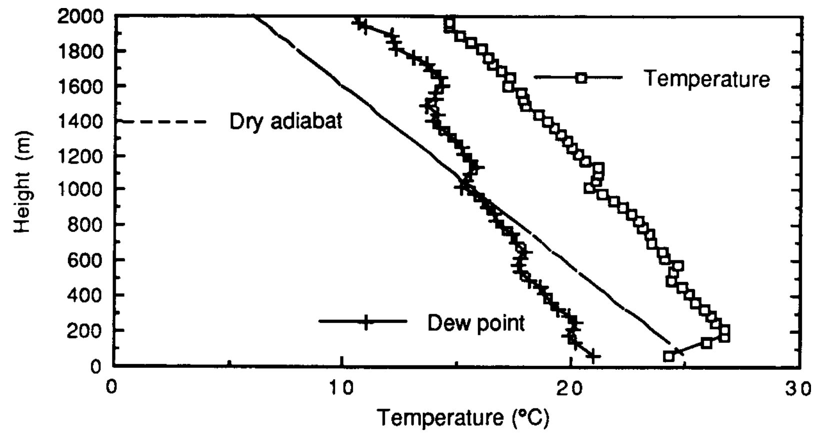

Fig 12 MATS #37: Temperature–dewpoint sounding from an Airsonde release made near the Weather Center Analysis Laboratory on Aug. 8, 1986, beginning at 0530 UTC (UTC differs from local standard time by 5 hours).

puff's travel speed and 215° for its direction (Fig. 13). The release of SF_6 began at 0700 UTC and lasted 15 minutes. The tracer cloud was first detected at 0751 to 0757 UTC while the TRAC vehicle was traveling toward its destination on the MATS arc 11.2 km downwind on SC 278 (just above "SRS" in Fig. 13), and a complete traverse across the cloud was completed. A puff released at 0715 UTC would have been transported downwind to within 2 km of the first encounter's peak concentration (assuming uniform winds at the height of release from about 205°). It was likely in this instance that the TRAC vehicle sampled the tail of the cloud, which had spread longitudinally and laterally.

As TRAC traveled farther toward the predicted crossing on the MATS arc, a second encounter with the tracer cloud occurred (at 0810 UTC) 24 km downwind near Windsor, S.C. Because of the time the cloud was detected and the mean wind speeds observed (5.9 m s^{-1}), the tracer sampled by TRAC during this traverse was estimated to have originated near the beginning of the release.

A puff trajectory for the second encounter was constructed on the basis of a release time of 0700 UTC and a travel time of about 70 minutes. The (assumed uniform) winds had shifted slightly more from the southwest by the time the puff reached the location near Windsor (as shown by the intermediate-length puff trajectory in Fig. 13). The locations of the measurements of the tracer cloud and the computed puff trajectory based on the assumption of uniform winds were within 2 to 3 km (as was true for the first encounter).

The third and subsequent encounters with the tracer cloud all occurred on the MATS arc. The third encounter was made with a puff that had been in transit for about 1.75 hours. Figure 13 shows the complete puff trajectory for the third encounter constructed with H area winds beginning at 0700 UTC (updated every 15 minutes as the puff moved downwind) and the locations on the MATS arc where tracer was detected. Those and subsequent crossings of the tracer cloud by the TRAC vehicle on the MATS arc showed that the MOS prediction and the measured positions of the peaks of the distributions were generally within 4 to 7 km.

Although many complexities in the wind field need to be considered for any particular experiment, the differences between puff trajectories and measured positions in this case help show the usefulness of assuming uniform transport winds at the release level on the basis of H area's measurements. Because of wind shear, the use of either higher or lower level winds would have led to significant errors in the predicted puff trajectory.

The TRAC vehicle stayed on the MATS arc for an additional 2.5 hours until 1100 UTC when sampling was terminated. The TRAC vehicle made a total of 16 traverses back and forth through the SF_6 cloud; each traverse took an average of 7 minutes to complete and covered a distance of 6 to 7 km along the highway. A few of the time cross sections of SF_6 (made roughly every 25 minutes by the TRAC vehicle) are shown in Fig. 14.

The model prediction for maximum concentration for the tracer cloud measured on the MATS arc was for 5000 to 6000 ppt of SF_6. The measurements show only a few hundred ppt during the several crossings. An investigation was made to try to determine why the model overpredicted by such an amount. Some possible reasons for the overprediction could have been large wind gusts causing dilution, wind shear causing increased tracer cloud width, and unusually large plume momentum rise (carrying the tracer cloud above the assumed equilibrium

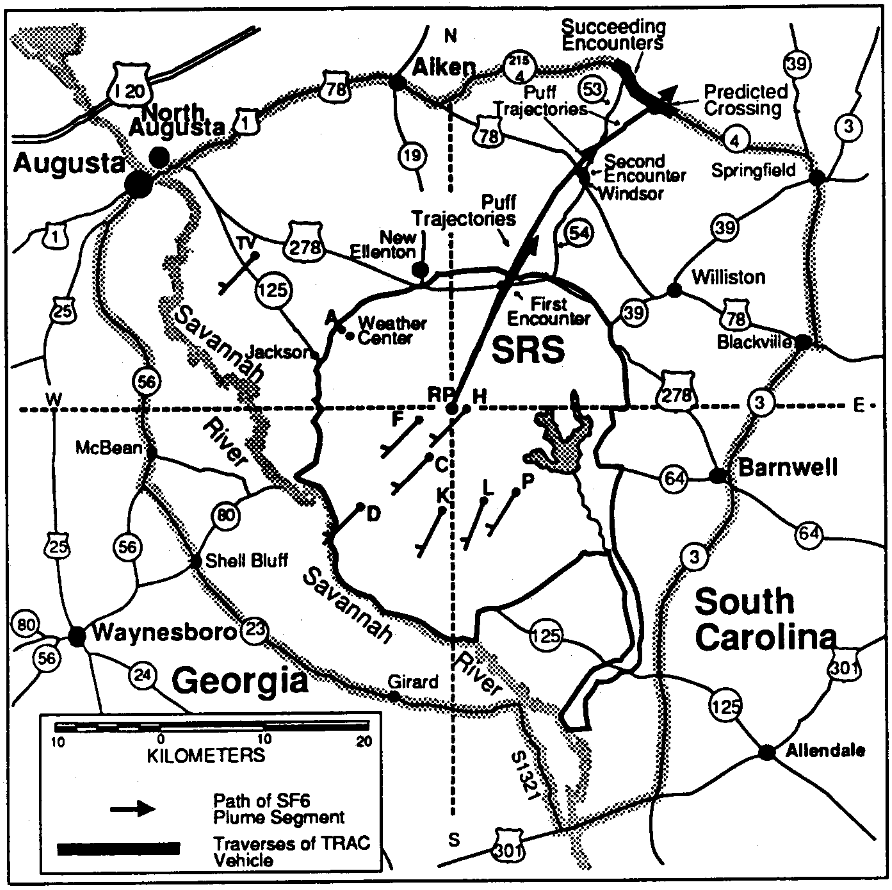

Fig 13 MATS #37: Experimental conditions on Aug. 8, 1986, showing 15-min average winds ($m\ s^{-1}$) from the tower network at the time the forecast for sampling teams was issued (0515 to 0529 UTC). The predicted crossing of the tracer cloud with the MATS arc is shown. Three puff trajectories based on 15-min-averaged winds measured at H area are also shown; the first is for winds beginning at 0715 UTC, the second is for winds beginning at 0700 UTC, the third also for winds beginning at 0700 UTC, but the tracer cloud is transported out to the MATS arc. Encounters where the continuous gas chromatograph aboard the TRAC vehicle measured the tracer cloud are also shown (UTC differs from local standard time by 5 hours).

level). None of these factors, however, seems sufficient to account for the magnitude of the observed differences. One additional possible reason for the overprediction could have been retention of downward diffusing tracer within the relatively slow-moving air beneath the forest canopy. This factor might also help explain the very large number of times that the TRAC vehicle was able to measure the cloud. An ongoing model evaluation study with a more complex model should eventually explain this overprediction.

SUMMARY AND CONCLUSIONS

The MATS experiments provide a dispersion data base and supporting meteorological data to distances up to 30 km downwind for short-term releases. The MATS program has met the objective of providing a data base that can be used to test emergency response dispersion models at SRS. The MATS program also provided a valuable means to test emergency response capabilities, such as logistics, communications, forecasting skills, and sampling strategies (although these items are not specifically addressed in this paper). The data continue to be used to help improve the understanding of dispersion at SRS as evaluations of more complex mesoscale models are ongoing.

Thirty-one MATS experiments were performed in the daytime with 15-minute releases of SF_6. The combination of 15-minute releases and 30-km downwind sampling is somewhat unusual for published dispersion experiments.

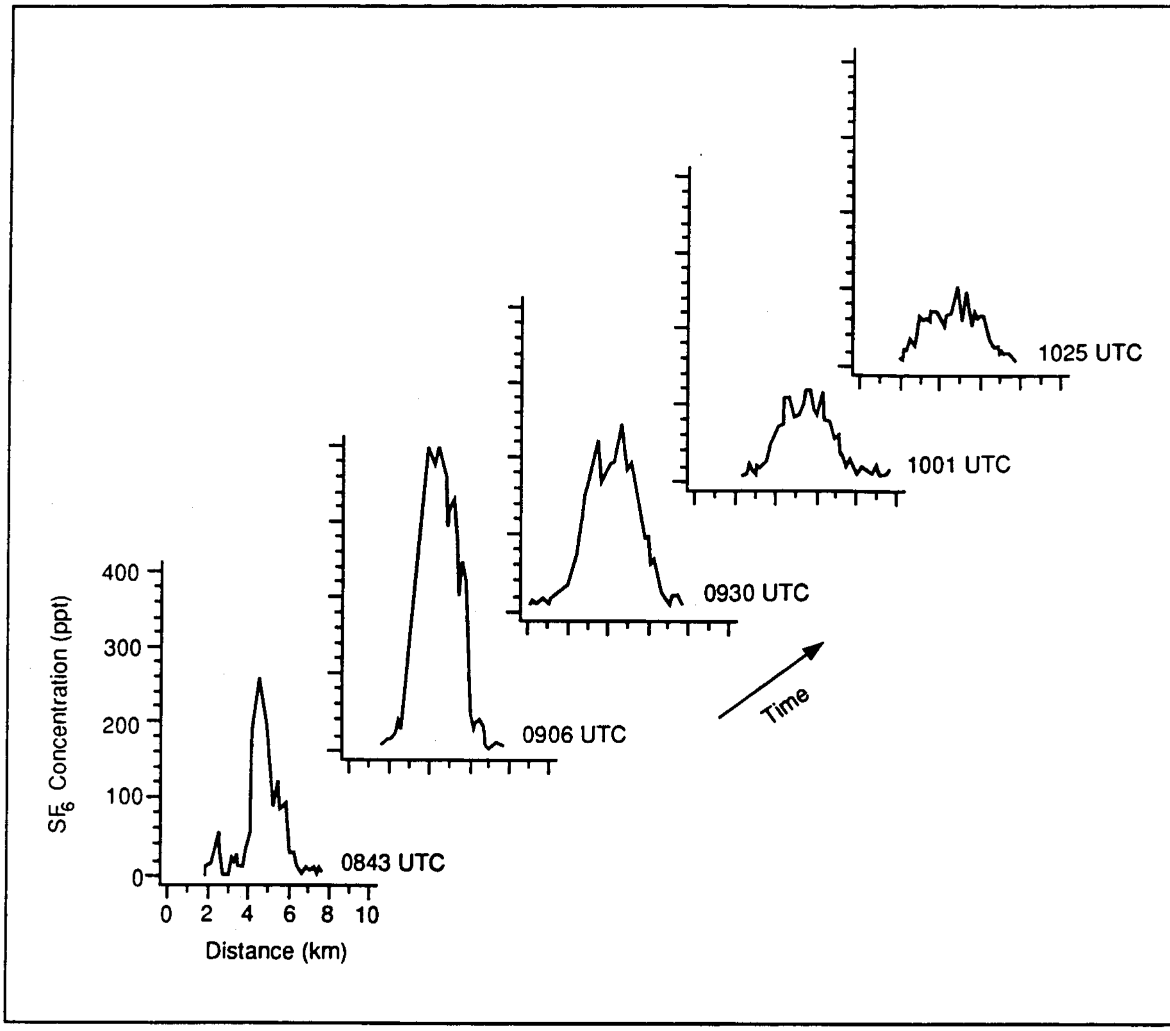

Fig 14 MATS #37: SF_6 concentrations measured by the TRAC vehicle on five separate traverses across the plume roughly 25 min apart.

Under typical atmospheric conditions during these experiments, the resulting tracer cloud was in an intermediate state between an idealized puff and a continuous plume. Such releases, however, are not uncommon at industrial installations and should be of interest to the dispersion community for this reason. The four nighttime experiments are of interest because of a general lack of nighttime dispersion data. The MATS experiments were conducted over heterogeneous terrain under a variety of synoptic conditions in all seasons of the year.

The two cases examined here illustrate the types of analyses possible with the MATS data. These cases show that the downwind locations of tracer material were adequately predicted if uniform winds at release level were assumed, but there are undoubtedly cases in the data set where such a simplification will not work. The amount of tracer during the daytime experiment was overpredicted by a factor of 2.5 of the measured amount. The concentration of the nighttime experiment was also overpredicted, but by an order of magnitude. Such discrepancies are not so unusual for mesoscale dispersion model predictions;[12] however, an effort is continuing at SRL to develop a model with more accurate representations of the complex atmospheric dispersion processes to better account for the observed tracer dilution.

The MATS program also improved aspects of field operations, such as sampler positioning, communications, and tracer analysis. This benefited the field program of the Stable Atmospheric Boundary Layer Experiment (STABLE) in April 1988 (Ref. 21).

The MATS data set can be used for diffusion research and modeling. It should provide a useful resource to the dispersion community for evaluating mesoscale models over gently rolling terrain.

ACKNOWLEDGMENTS

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Book Review: *Environmental Radioactivity in the European Community 1984–1985–1986*[a]

By C. A. Little[b]

This report is a compilation of environmental radioactivity data based on information provided by member European Community states. The report is organized into six chapters: Airborne Particulates, Deposition, River Water, Drinking Water, Milk, and Chernobyl. Generally speaking, data are reported as national averages; the exception is data for the larger nations—Germany, France, Italy, and the United Kingdom—which are reported as three or four regional values for each country. For example, Germany is partitioned into North, Central, South, and West Berlin.

Each of the media chapters consists of a discussion/introduction that provides "general information on the medium in question and the occurrence of natural radionuclides therein, together with a short description of sample preparation and measuring methods." Each chapter also contains maps showing the location of samples for each country or portion of a country. The majority of the report consists of tabulations by country, year, and quarter of the nuclides and media sampled. Radioactivity reported for each of the sampled media are as listed in Table 1.

Table 1 Nuclides Reported by Categories of Data

Chapter	Nuclides reported
Airborne Particulates	Total α, total β, ^{137}Cs, ^{90}Sr
Deposition	Total β, ^{137}Cs, ^{90}Sr, ^{3}H
River Water	Total α, residual β, ^{137}Cs, ^{90}Sr, ^{3}H
Drinking Water	Total α, total β, ^{137}Cs, ^{90}Sr, ^{3}H
Milk	β(Sr + rare earths), ^{137}Cs, ^{90}Sr

The chapter on Chernobyl is organized differently. For air particulate concentrations of ^{131}I and ^{137}Cs, air concentration profiles through time are given for 13 different locations to provide a spatial picture of how the contamination spread. A series of six maps illustrates the spread of ^{137}Cs during the Chernobyl event; for up to 60 European locations average daily concentrations of ^{137}Cs in air are plotted for each 24-hour period between 9:00 a.m. on April 29 and 9:00 a.m. on May 5, 1986. In addition, total ^{137}Cs deposition by country is tabulated.

Several useful appendices to the report include Methods of Calculating Time and Geographical Averages (Appendix A), Addresses of Competent National Authorities (Appendix B), and a bibliography (Appendix C).

A word of warning: There are a few errors of labeling in the report. One figure purports to give air concentrations between April 27th and April 15th; they really mean May 15th. For the Chernobyl chapter, the list of figure captions in Appendix E states that "I-137" concentrations are shown.

Despite these niggling errors, this report may be useful to those with interest in environmental transport or modeling. At the least, it serves as an archive of data for the period.

[a]Radiation Protection Report No. 46, Joint Research Centre, Commission of the European Communities, Luxembourg.

[b]Health and Safety Research Division, Oak Ridge National Laboratory, Grand Junction, Colo.

Waste and Spent Fuel Management

Edited by K. J. Notz, Jr.

Activities Related to Waste and Spent Fuel Management

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

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

The information in this issue of *Nuclear Safety* was received during July, August, and September 1991.

NRC's BRC POLICY STILL UNDER ATTACK

The Below Regulatory Concern (BRC) policy established by the Nuclear Regulatory Commission (NRC) establishes a framework for making decisions on granting exemptions from the Commission regulations dealing with waste management and disposal in cases where radiation levels are "so low that they do not require the imposition of regulatory controls to ensure protection of public health and safety."[1] Public reaction to the BRC policy appears to be similar to that stated by one senator: "what may be below the concern of NRC is of tremendous concern to many others."

Two aspects of the continuing concern about this issue are discussed in the following text.

NRC Declares Moratorium on BRC Policy Implementation

The NRC announced in late June 1991 that it was declaring a moratorium on the implementation of the BRC Policy. In lieu of implementation of the policy, the Commission approved the initiation of a phased consensus-building process on BRC issues.[2]

As reported in the last issue of *Nuclear Safety*,[3] F. X. Cameron, Deputy Administrator for the Office of Licensing Support Systems at NRC, and H. Bellman, an independent arbitrator with experience in consensus building, began collecting information and performing their evaluation of a broad consensus-building process to develop a base of understanding and support for the NRC's BRC Policy. Cameron and Bellman concluded that a consensus process relating to BRC is feasible. Cameron stated that "there seems to be a broad enough base of support for such a process among the groups that we interviewed" and noted that "the primary objective of the process would be the provision of advice by a consensus body to the Commission on the entire range of BRC issues."

The Commission believes that, for the consensus-building process to be effective as a forum to evaluate the entire range of issues related to the BRC policy, it is essential that representatives of all affected parties, especially those groups who have demonstrated a major interest in the policy, participate in the process. The Commission established a target date of December 1992 as the closure date for the consensus body to provide its final advice.

Regulatory responsibilities of NRC were to continue during the consensus-building process. All activities necessary to provide adequate protection of public health and safety and the environment, particularly those activities

[a]Oak Ridge National Laboratory.

concerned with the cleanup of contaminated sites in a timely manner, were to be unaffected. Therefore, although the Commission deferred the implementation of the BRC policy, it made clear that it intended to continue to address issues related to waste disposal, consumer products, recycling of materials, and decontamination and decommissioning, to the extent necessary and on a case-by-case basis, in the same manner in which these issues were considered before the publication of the BRC policy statement.

Accordingly, the NRC staff planned to continue to make licensing decisions involving exemptions or site decommissioning, using existing rules, criteria, and practices, and intended to inform the Commission of all significant or controversial actions of this type.

States Argue for Right to Decide on BRC

In 1990 the House of Representatives' Interior Committee unanimously passed a bill allowing states greater sovereignty in determining whether or not to accept low-level radioactive waste designated as BRC by the NRC, but the congressional session concluded before the bill could be voted on by the full House. Therefore the Chairman of the Interior Committee, Rep. G. Miller (D-Calif.), reintroduced legislation (H.R. 645) that would (1) permit state regulation of radioactive waste that NRC does not require to be disposed of in a licensed repository and (2) revoke the NRC's 1990 BRC policy statement, which sought to define radiation levels the agency believes pose such a small safety risk that further regulation of these items would be an inefficient use of agency resources.[4] [Senate Majority Leader G. J. Mitchell (D-Maine) introduced S.1111 in the Senate in response to the NRC's BRC Policy Statement. The Senate had yet to hold a hearing on the measure.[5]]

In the first hearing on H.R. 645, Rep. P. H. Kostmayer (D-Pa.), chair of the Subcommittee on Energy and the Environment, heard testimony from five separate panels of witnesses, all of whom, with the exception of the NRC representative, applauded the intent of the bill to revoke the NRC's BRC policy and urged that even stronger legislative measures be taken to ban the deregulation of BRC waste.

The Attorneys General of the states of Vermont, Ohio, and New York also voiced their support of H.R. 645 and strongly criticized NRC for its BRC policy. The state officials collectively noted that NRC neglected to conduct any formal rulemaking in establishing the policy "nor did it perform any environmental or health impact statement." The officials added that, "Although the NRC stated that the policy would adequately protect the public health, it also clearly stated that the impetus for the policy is reduced costs."

Dr. J. H. Johnsrud, an expert on the health effects of radiation, argued in short that: "When all sources of doses from naturally-occurring radiation plus permissible releases and proposed BRC exposures are combined, even one millirem of additional exposure may be an unacceptable threat to public health."

Another expert who testified on the health effects of ionizing radiation was Dr. G. T. Davis. Davis proclaimed that NRC has underestimated "by a factor of four" the number of excess cancers that exposure to low-level radioactive wastes could cause. Davis disagrees with the methodology used by NRC to arrive at its conclusions on the minimal adverse health affects of BRC. Said Davis: "failure to use appropriate dose-response predictions, inappropriate truncation of low doses, use of the threshold hypothesis, and failure to account for alpha radiation leads to even greater underestimation and minimization of the effects of BRC implementation."

The Commission was represented at the hearing by F. X. Cameron, who developed the consensus-building approach, and by R. M. Bernero, Director of the Office of Nuclear Materials and Safeguards. Cameron was quick to admit that there was almost unanimous dissatisfaction with the manner in which NRC developed its BRC policy. But, on the basis of his discussions with affected parties, he maintained that a consensus-building process, in which everyone participates, could result in a more acceptable BRC policy.

Kostmayer was clearly unconvinced by the NRC position and asked Cameron how he could believe that it is only the *process* that is unsatisfactory when ten states have already gone so far as to enact their own laws regarding BRC. Cameron agreed that in those cases there must also be dissatisfaction with the substance of the policy.

At that point Bernero stepped in to try to explain to a bewildered Kostmayer why NRC and the Environmental Protection Agency, and several other radiation regulatory agencies, differed in their outlooks on what constitutes a BRC radiation level. Bernero said that NRC allows individual, per-year dose exposures of up to 10 millirem for a narrow population exposure standard. That is, where very small groups are exposed to only one source of radiation, a higher level of exposure would be acceptable. For groups of greater number, however, who are exposed to more than one source of radiation, the level considered safe is lower, said Bernero. The EPA differs on the maximum exposure level permissible, maintaining that 4 millirem per year per individual should be the highest allowed. Bernero explained that the difference between levels was, radiologically speaking, very slight.

Except for the NRC contingent, every witness at the hearing supported passage of H.R. 645, but almost every witness also suggested that the bill should go further. New York's Attorney General, represented by J. M. Marous, suggested a strengthening amendment that would allow states to regulate *all* low-level waste in the state. Citizen enforcement and supervision of low-level waste sites should also be implemented to build public trust, said Marous.

The NRC still needed to go through a formal rulemaking process before it could put its BRC policy into effect. In early July 1991, NRC announced that it would delay its new policy and negotiate the issue with all interested parties if opponents promised to give up their rights to seek intervention by the courts or Congress. Environmental groups had yet to embrace the offer.

FIRST STUDIES COMMENCE AT YUCCA MOUNTAIN

In 1982, Congress passed the Nuclear Waste Policy Act, which, along with its amendments, provides the framework for the nation's program for the disposal of civilian high-level radioactive waste (HLW). The Department of Energy (DOE) is charged with managing the permanent disposal of HLW. The 1987 amendments of the Nuclear Waste Policy Act designated a site at Yucca Mountain, Nev., as the sole site to be characterized for potential suitability as a mined geologic repository. The amendments also created the Nuclear Waste Technical Review Board (NWTRB) to evaluate the scientific and technical validity of DOE's activities to characterize Yucca Mountain as a potential site for such a repository as well as activities related to packaging and transporting high-level waste.

Activities on a number of aspects of this program are briefly discussed.

Nevada Reluctantly Issues Air-Quality, Underground Injection Permits to DOE

After years of disagreement between the state of Nevada and DOE that attended DOE's efforts to obtain the permits needed for scientists to perform site characterization tests at what could be the nation's first high-level radioactive waste repository, Nevada finally approved two of the three permits DOE considered most crucial to enabling studies to progress.[6]

An air-quality permit, which the DOE filed in June 1988, was finally approved by the state of Nevada in mid-June 1991. A Federal Judge in Las Vegas had earlier threatened to force the state to review the permit application in a timely manner. Then in mid-July 1991, Nevada approved another of DOE's most important permits, an underground injection control permit. DOE originally filed that permit with Nevada in April 1989.

Citing the progress that is being made on the third permit application for groundwater appropriation, Nevada moved that DOE's entire court complaint (seeking court-ordered final action on all the applications) be dismissed. Although DOE did not oppose dismissal of the part of the case that pertained to the two permits already issued, the Department did argue that the court should retain jurisdiction over the water appropriation matter until final action had been taken by the state. Federal Judge H. D. McKibben did grant dismissal of the part of the case concerning the two permits already processed but ordered that another "status" hearing on the state's progress in processing the water permit would be held in November 1991. Although not forcing the state to take final action by that time, McKibben indicated that his time frame should allow "reasonable time" for the action to be taken.

C. Gertz, the Yucca Mountain Project Manager, and Dr. J. Bartlett, Director of the Office of Civilian Radioactive Waste Management (OCRWM) at DOE, took turns commenting on the work at Yucca Mountain and fielding questions from reporters. Gertz, who has been working at Yucca for 4 years, said, "This is the biggest milestone I've seen yet. It's a small but significant step, and it marks the beginning of the comprehensive scientific investigations." Gertz described the new surface-disturbing scientific work set to take place in July 1991: trenching at Midway Valley to study existing and potential faults, deepening Trench 14 to examine the origin of calcite–silica deposits, and excavating pits to study the age and origin of volcanoes.

Gertz highlighted several areas of uncertainty his team still faced. He stressed several times during the briefing the extreme importance of obtaining approval of the water permit from the state of Nevada. Water is crucial to carrying out the surface-disturbing activities; the state will not allow disturbance of the soil unless dust is controlled during excavations, drilling, and cutting. Currently, said Gertz, water is hauled 45 miles from Death Valley, Calif., at great expense to the project. The water the project needs is accessible no more than half a mile away from the site. Moreover, Gertz stated that the requested water is approximately 2% of the annual use of one local mine. Other uncertainties involve the state's action on pending and future permits (a total of 18 are required by DOE for various stages of site characterization) and Yucca's annual budget, described by Gertz and Bartlett as "limited."

Bartlett was asked why he considered the $170 million budgeted for activities in FY 1992 a "limited" amount of funds. Bartlett replied: "That's a fair question. . . ." Gertz promptly jumped in with: "It seems like a lot. But when you consider that it costs $12 million just to keep one of our drills operating round the clock for a few months, and we will require several of those drills, you begin to get a better picture of the constraints we're under."

On licensing, Gertz noted that Yucca Mountain was in a pre-application stage and maintained a constant dialogue with NRC. Gertz stated that, assuming the site were to be positively characterized as a safe location for a repository and that the project stayed on track, the license application would be made in 2001. At that time the NRC would have 3 years to approve or deny DOE's license application. If the NRC then approves it, the license would be good for 25 years.

Gertz and Bartlett made it very clear that they strongly desire federal legislation to facilitate the timely processing of their permits. In the Senate, a bill was recently reported out by the Senate Energy Committee directing DOE to carry out site characterization activities with all reasonable speed. The Senate's bill would prohibit Nevada from issuing injunctions delaying characterization activities.

Meanwhile the House Energy and Commerce Subcommittee on Energy and Power approved by voice-vote in early September 1991 a provision in a draft energy strategy bill that would allow the DOE to move forward with testing the suitability of the Yucca Mountain site in Nevada for disposal of high-level waste.[7] Both the House and the Senate bills were designed to get around Nevada's attempts to block the repository, which would accept highly radioactive waste from spent fuel from the nation's nuclear power plants. Nevada has been unwavering in its opposition to the use of Yucca Mountain as an HLW repository ever since it was decided in 1987 that Yucca Mountain would be the only site for consideration for the repository. Since that time the state government has refused to issue a series of state permits required for geologic study of the site.

The measure approved by the House would allow the Federal Government to begin work on the site without state approval. Still other state permits would be required for subsequent work. Rep. M. Synar (D-Okla.) warned his colleagues that "this (stripping states of their rights) could happen to your states on some bill in the future."

NRC Addresses EPA's HLW Waste Standards for Repository Licensing

In mid-August 1991 the NRC staff members met with the Commissioners to address "uncertainties" in implementing the EPA's high-level waste standards.[8] The NRC staff had a number of important issues to brief the Commissioners on regarding how EPA's HLW standard should be applied to the licensing of an HLW repository, such as the site under investigation at Yucca Mountain in Nevada.

J. M. Taylor, Executive Director for Operations, summarized the staff's views on dealing with uncertainties in EPA's probabilistic HLW standards. Taylor first referred to a 1986 staff paper on licensing a geologic repository that described ways to streamline the hearing process, identify and resolve licensing issues early, and improve appeal processes. The 1986 paper was a foundation for the staff's regulatory framework and was updated in 1990 as SECY-90-207. That paper described the existing regulatory framework for licensing a repository, approaches for identifying uncertainties, and a strategy and schedule for reducing uncertainties by using a mix of rulemakings, technical positions, and regulatory guides. Taylor continued to say that another 1989 staff paper informed the Commission that EPA's 1985 standards had been remanded to EPA for further analysis in a 1987 court decision and of the status of EPA's efforts to reissue its HLW disposal standards. It also stated that the NRC staff had reevaluated its views on implementation of probabilistic standards and that the staff recommended development of procedures and rules needed for implementing the standards.

Taylor also described the staff's latest analytic efforts regarding EPA's standards for a repository and noted that the current paper distinguishes between "regulatory uncertainties" and "technical uncertainties." Regulatory uncertainties involve what must be demonstrated to show compliance with EPA's standards and can be reduced or eliminated through wording, whereas technical uncertainties are inherent in repository performance assessments and will be encountered in implementing EPA's standards regardless of the form those standards might ultimately take. The technical uncertainties can be categorized as data uncertainties, future states uncertainties, and model uncertainties. Although Taylor made a point to say that a methodology to deal with regulatory uncertainties has been "reasonably well-developed," he noted that the same is not true for dealing with technical uncertainties. The technical process is rather an "on-going, iterative" function. As such, development of methodology for reducing technical uncertainties is not complete until the uncertainties in repository performance have been reduced to an acceptable level, said Taylor. He also noted that dealing with technical uncertainties is primarily the responsibility of the DOE and that the NRC's role in site characterization is really one of preapplication consultation, per the Nuclear Waste Policy Act.

Even after those issues are addressed, D. Fehringer of the High Level Waste Management Office pointed out that some residual uncertainties will remain as the result of the long time period of concern, the multiple science disciplines involved, and the need to consider natural and human disruptions.

What Site Characterization Entails

Site characterization of Yucca Mountain focuses on three key issues: volcanoes, earthquakes, and the time it takes water to percolate through the mountain. The DOE needs enough information on these phenomena to understand the complex geology and hydrology of the site. Only with this information can DOE hope to predict what is likely to happen 10 millennia from now. That's the length of time, EPA has said, that must elapse before any radionuclides can be permitted to reach the environment from a repository.

Excerpts from an article that provides details of what site characterization entails, written by Alice Clamp and published in *Nuclear Industry*,[9] are as follows:[a]

> Standing on top of Yucca Mountain, the world of commerce and industry seems far away. To the north, barren ridges alternate with dusty valleys all the way to the horizon. To the south lies the Amargosa Desert and beyond a mountain range, Death Valley.
>
> The mountain seems like a good place to bury high-level radioactive waste. It's isolated. The climate is arid. But it's what you can't see that counts. That's why hundreds of scientists will spend billions of dollars and about 10 years studying the inner workings of Yucca Mountain, to see if the site is suitable as a repository for spent fuel from U.S. nuclear power plants.
>
> That scrutiny is termed site characterization—the detailed investigation of Yucca Mountain's geology, hydrology and geochemistry. . . . Site characterization is a massive undertaking, possibly the most comprehensive. . . assessment of a chunk of real estate ever conducted. It will entail 329 boreholes ranging in depth from five feet to 5,000 feet, scores of trenches, between seven and 10 miles of tunnels, and thousands of tests—plate loading, rock mass strength, percolation, hydrochemistry, diffusion and heated block, to name but a few. It will require up to 10 years of on-site and laboratory investigations and analyses, cost more than $4 billion and involve some 500 scientists and engineers, not to mention support personnel and managers. In some areas, it's a first-of-a-kind effort, requiring novel research methods and adaptation of monitoring and measuring equipment. Little wonder that the project has attracted controversy, both political and scientific.
>
> Complex as it is, the characterization focuses on three key issues: volcanoes, earthquakes and the time it takes water to percolate through the mountain. . . .
>
> VOLCANOES. Among other questions, site characterization must show whether volcanoes could be a threat to a repository. All the evidence so far suggests they will not. . . . The area around Yucca Mountain has seen only five cases of volcanic activity in the last 4 million years. To the west, there's a 3.7-million-year-old cluster of volcanic cones in Crater Flat and another, younger cluster, about 1 million to 1.2 million years old. To the northwest, beyond the horizon, there's a 2.8 million-year-old cluster in Buckboard Mesa and beyond that, two more volcanic cones. . . about 300,000 years old. Finally, there's Lathrop Wells to the southwest.
>
> Lathrop Wells is anywhere between 20,000 and 150,000 years old, says B. Crowe, an expert in volcanism from Los Alamos National Laboratory. Once he gets permission from the state of Nevada to dig a trench at the site, Crowe says, he'll know the volcano's age in three to six months. How? By dating soil samples using a method called thermoluminescence, or TL. From the time a volcano erupts, its TL clock starts ticking as elements in the soil give off gamma particles. Crowe is confident of TL's accuracy. "We are testing it on soils all over the West that were dated using the carbon-14 method. So far, there have been no discrepancies."
>
> Crowe and his colleagues. . . also plan to dig trenches at the volcanic centers that are less than half a million years old, to verify their age. And they'll drill boreholes at the three volcanic clusters in Crater Flat, taking samples and dating them using the potassium–argon method. Volcanologists need to know the age of all five volcanic clusters so they can calculate the recurrence rate—the time between eruptions. That information will tell them if and when another volcanic event could [be expected to] occur.
>
> They're also trying to answer other questions. What is the line, or lines, along which these old volcanoes erupted?

[a]*Editor's Note: Nuclear Safety* does not often publish industry-sponsored material without independent peer review to assure the objectivity and freedom from bias of the text. In this instance we have chosen to do so as a way of informing our readers about a subject somewhat removed from the expertise of most of them. Please note that this excerpt has not been peer-reviewed.

In what direction was the magma... dispersed when it erupted from the volcano? That information will help them determine the likelihood that an eruption would disrupt a repository, and if it did, the likelihood that it would bring radionuclides to the surface. . . .

EARTHQUAKES. Major volcanic eruptions haven't occurred in the area for tens of thousands of years. But minor ground movement—below 1 on the Richter scale—happens quite often. "It's imperceptible without sensitive monitoring equipment," says the USGS's [U.S. Geological Survey] A. Buono. On a map, he points out the location of 55 monitoring stations, now being upgraded so that seismologists can pinpoint active faults and measure earthquake centers even more accurately.

The USGS has identified 32 faults in the area. Their names summon up images of the old West: Stage Coach Road, Solitario Canyon, Windy Wash, Bow Ridge. Most of the faults lie to the west of Yucca Mountain, but one—the Ghost Dance—runs right through the mountain. "The Ghost Dance fault has moved little compared with the others—between 50 feet and 150 feet vertically," says Buono. "And based on our studies, most of that probably happened 11-12 million years ago."

To find out just how much these faults have moved, and when, seismologists will take a close-up look. They've already examined the soil and rock beds across the major faults using trenches dug in the 1980s, before the state required permits for such surface disturbing work. That's how they know that the Windy Wash fault—about two miles west of Yucca Mountain—has moved a minuscule foot-and-a-half vertically in the last two million years. The movement took place in seven episodes, the last occurring less than 10,000 years ago.

Seismologists also keep their finger on the earth's pulse, monitoring the minor ground movement occurring at the mountain today. A network of geodetic leveling stations tells them of any movement—vertical or horizontal.

As part of site characterization, geologists will dig trenches 6-20 feet deep at various spots along most of the faults. They'll also dig tunnels, or drifts, across the faults in the mountain to look for any movement, especially in the last 10,000 years. "Knowing the magnitude and date of movement of the faults in the area can help us predict what's likely to happen over the next 10,000 years," says the USGS's Buono.

GROUND WATER. Scientists around the world have studied earthquakes and volcanoes, so there's really nothing new about the tweaking and poking of these phenomena at Yucca Mountain. What is new is the examination of how water moves through the earth's "unsaturated zone"—an area of rock containing very little water. "The rock is dry in appearance, but there's some moisture in it, just like there's humidity in the air," says C. Gertz, who heads DOE's Yucca Mountain Project Office in Las Vegas.

At Yucca Mountain, the unsaturated zone is large, extending from the surface to a depth of about 1,800 feet. Below the unsaturated zone is the saturated zone, where the water table begins. In the saturated zone, ground water moves horizontally and, after about five miles, reaches the environment. A potential repository would be sited about 1,000 feet beneath the mountain's surface and about 800 feet above the water table.

DOE needs to know about the movement of water in the mountain: how much precipitation gets into the rock, how far and how fast it travels, and how the rock's structure and properties affect its movement. That information, gathered by the project's geohydrologists, geophysicists, geochemists and hydrochemists, will allow assessments of whether moisture could reach a repository and—assuming some release of radionuclides from the repository—then reach the water table. . . .

Water first enters the mountain as precipitation, says A. Flint, a USGS scientist in charge of the hydrologic investigation program. That's why he needs to measure the amount of rainfall or snowfall at the site. But without permits from the state, Flint can't build the concrete pads required for installing sophisticated rain gauges. To get some idea of precipitation in the meantime, he has improvised. "We bought $10 plastic rain wedges and put them on two-by-fours that we then strapped to the steel casings of boreholes drilled in the mid-1980s, before the state said permits were needed." Yucca Mountain gets only about six inches of precipitation a year, so Flint didn't want to miss even a drop.

What happens to the rain or snow once it hits the ground? Using boreholes, Flint and his colleagues have traced it to a depth of about six feet at various locations on the mountain. Most of it, "maybe even 100 percent," returns to the atmosphere through evaporation or plant transpiration... says Flint.

All the same, he's analyzing the soil to determine if any water does travel too far down to be evaporated. By figuring out the age of the water in the unsaturated zone, scientists can learn about the rock's ability to retain or transmit water.

Using existing boreholes, USGS scientists have used neutron moisture meters to track water movement down to 150 feet below the surface. Site characterization entails more extensive and sophisticated monitoring. "We'll have instrument 'stations' at various depths in new boreholes," says the USGS's Buono. The stations—strings of equipment designed to monitor changes in moisture content—will operate in the top 50 feet of the unsaturated zone (the shallow infiltration program) and from there down to nearly 2,500 feet (the deep percolation program). The stations will remain in place for three to five years, sending data to a computer system on the surface for processing.

Besides wanting to know how much water is in the rock, scientists want to know the water's makeup and age. So, hydrochemists will take water samples from the borehole stations to analyze in the laboratory. Besides water chemistry tests, they'll date the water. "If you get levels of tritium at great depths in the rock that are consistent with those in the atmosphere as a result of above-ground atomic bomb tests in the 1940s, '50s and '60s, that means water has migrated pretty quickly," says USGS project officer Tony Buono. "On the other hand, if you find through carbon-14 dating that the water 1,000 feet down is 40,000 years old, that tells us the unsaturated zone greatly inhibits water movement."

NATURAL BARRIERS. Scientists will do more than study how water travels through the rock of the unsaturated zone, however. What's true for the movement of water isn't necessarily true for the movement of dissolved radionuclides. Geochemists from Los Alamos National Laboratory are looking at minerals in the rock that retard the movement of specific radionuclides. Of special interest, explains R. Herbst, an engineer and Los Alamos project officer, is zeolite, a mineral in relative abundance throughout the 1,800 foot-thick unsaturated zone.

Zeolites have the ability to swap ions... with radionuclides. As a result, they capture the radioactive components and hold them in the rock while they decay. But zeolites aren't equally effective with all types of radionuclides. "Our research is aimed at sorting out which components are retarded and how much, and learning whether there are any that aren't retarded at all," says Herbst.

Zeolites aren't the only natural barrier to the movement of radionuclides, so the scientists working with Herbst have used borehole samples to identify all the minerals in the mountain and their vertical distribution. To learn about the lateral distribution of minerals, they'll need permits from Nevada to dig additional boreholes.

Scientists from Los Alamos will also conduct studies in the underground tunnels. They'll spend five to 10 years tracking tracer-laden water that has been allowed to infiltrate naturally into the unsaturated zone beneath the level of a potential repository. Water samples will be analyzed for the presence of various tracers—"stand-ins" for actual radionuclides—and water movement will be followed using neutron probes. Scientists will use this information to estimate how long it would take water to travel from a repository to the environment... .

I. Winograd, a research hydrologist with the USGS, suggests that the archaeological record dating back more than 40,000 years demonstrates how well both durable and fragile items have survived, even in saturated zones. One example, he says, is the mummified plant fossils found in 10,000-45,000-year-old pack rat middens—refuse heaps—throughout arid and semiarid regions of the southwestern United States.

"If nature can do that without trying, just think what we can do if we apply ourselves," says D. Langmuir, a geochemist and member of the Nuclear Waste Technical Review Board.

GERMANY CONFRONTS WASTE ISSUE

Germany has recently begun confronting the problem of what to do with its high-level nuclear waste, in earnest. Although western Germany gets about one-third of its electricity from nuclear power plants, it has no permanent disposal site for the waste they produce. As the government pushes for the construction of new nuclear reactors in eastern Germany, the disposal question has come to the fore. The Konrad nuclear waste disposal site, located near Hanover in the state of Lower Saxony in northern Germany, is being explored as an option. The site is a former iron ore mine. A German environmental activist said that "Whether people accept the construction of new nuclear reactors depends very much on whether they've solved the problem of nuclear waste disposal."[10]

DOE AND SWITZERLAND SIGN AGREEMENT

Switzerland's national cooperative for the disposal of radioactive waste (NAGRA) signed an agreement with the U.S. DOE's OCRWM establishing a new, extended project agreement that deals with the geological, geophysical, geochemical, hydrological, and structural effects from a mined geologic radioactive waste repository. The NAGRA operates unique facilities not available in the United States; this gives DOE the opportunity to test critical instrumentation, computer models, and field methods according to DOE. The Department also said that the new project agreement will be very important to U.S. efforts at the Yucca Mountain project site since it will stress the effects of the flow of groundwater and radionuclide transport on a repository in a fractured rock formation.[11]

NRC STAFF NOTIFIES OFFICIALS IN 50 STATES OF SLIGHTLY CONTAMINATED FENCING BARS

The NRC staff has advised radiation control officials in all 50 states of the discovery of steel chain-link fencing parts (tension and gate bars and truss rods) imported from India that have been found to be slightly contaminated by radioactive cobalt-60 (Ref. 12). The components were imported by two U.S. companies: Transmark Sales of Riverside, Calif., and by Steel City USA, Inc. (Promet, Inc.) of Houston, Tex. The steel for the fencing parts was made by two Indian companies.

Efforts by NRC include contacting the Government of India to ensure that the Government is aware of the problem and to attempt to determine the scope and source of the contamination. Further, the NRC said the products found to have radiation levels above natural background will be separated out and either returned to India or handled as radioactive waste.

The NRC staff recommends that no action be taken for tension bars, gate bars, or truss rods already installed in fences or in the possession of retailers and fencing companies because of the low radiation levels of the contaminated products. The cobalt-60 is embedded in the steel and, under normal handling, people will not become contaminated, NRC said.

On the basis of actual measurements of the radioactive content of the steel made to date, NRC has calculated that radiation doses to members of the public or workers actually engaged in handling the material would be very small. Members of the public standing near the fencing bars or trusses for 8 hours a day, 365 days a year would receive an annual dose of much less than 1 millirem—compared with an annual exposure from natural background radiation of approximately 300 millirem per year and a current NRC limit of 500 millirem per year for exposures to members of the public from NRC-licensed activities. It is conservatively calculated that exposures to workers handling radioactive fencing material would not exceed about 40 millirem per year, according to NRC.

The NRC found out about the problem with the fencing parts after a fencing company truck was routinely surveyed for radioactivity as it was leaving a DOE facility in Hanford, Wash., on August 9. The radioactivity was found to result from two chain-link fence tension bars that contained cobalt-60. The components were traced to a U.S. distributor in La Habra, Calif.

Further investigation by NRC and various state governments showed that the radioactive fence parts were imported by two U.S. companies.

The NRC sent a letter in mid-August to importers and distributors who have been identified as possible or actual recipients of the contaminated fencing parts from India confirming actions that the companies have voluntarily agreed to take to avoid further distribution of contaminated fencing parts within the United States, according to NRC. The companies will be responsible for ensuring that all steel products in their inventory from India and any incoming shipments of steel from India are checked for radiation. The companies will also ensure proper handling of any contaminated products at their facilities.

IIT REPORTS ON POTENTIAL CRITICALITY IN A WASTE TREATMENT TANK AT THE GE NUCLEAR FUEL FACILITY

The NRC Incident Investigation Team (IIT) assigned to report on the May 28, 1991, potential criticality incident at the General Electric (GE) Company's Nuclear Fuel and Component Manufacturing (NFCM) facility in Wilmington, N.C., briefed the NRC in mid-September on the root causes of the incident and suggested corrective actions to prevent similar events.[13]

The incident occurred when an estimated 150 kilograms of uranium were inadvertently transferred to an "unfavorable geometry" waste treatment tank. "Unfavorable geometry" is a term used to designate a container or vessel that can hold fissile material, such as ^{235}U, in sufficient amount and in a sufficiently compact geometry so that criticality can potentially occur. The "double contingency principle,"[14] requires that, where such a vessel is used, at least one other unlikely, independent, and concurrent change in the process conditions (e.g., degree of moderation, density, or reflection) must have to occur before an accidental nuclear criticality is possible in such a vessel. According to the IIT, because of the tank configuration and type and quantity of material available, there was the potential for a nuclear criticality accident that would yield a burst of neutron and gamma radiation sufficiently intense to injure or kill persons located close to the event and cause radiation exposures at greater distances. Dispersion of radioactive contamination in the immediate vicinity would also be expected; however, there off-site radiological impacts would not be likely.

The NFCM facility is licensed by NRC to possess and use special nuclear material to be used for such processes as uranium hexafluoride conversion, fuel manufacturing, scrap recovery, process technology operations, laboratory operations, and waste treatment and disposal. A part of the fuel manufacturing facility is a uranium recycle unit to recover uranium from certain waste and scrap materials. In this process scrap materials are dissolved in nitric acid, passed through a filter, and fed to a solvent-extraction system. The recovered uranium is then returned to the fuel manufacturing process.

The team concluded that there were three interrelated "root causes" that contributed to the incident:[15]

1. There was a pervasive attitude within GE that nuclear criticality was not a credible accident scenario.
2. GE management did not provide effective guidance and oversight of its NRC-licensed activities to assure that operations were conducted in a safe manner.
3. There was a deep-seated, production-minded orientation within the organization that was not sufficiently tempered by a safety-first attitude, particularly regarding nuclear criticality safety.

The NRC maintains that, together, these three causes manifested themselves in contributing such causes as design deficiencies, procedural noncompliance, inadequate incident investigations, and a general deterioration of criticality safety. Because of these problems, there was little or no latitude in the licensee's program to accommodate equipment failures, system upsets, or personnel error.

The IIT also concluded, however, that there were deficiencies in NRC regulatory oversight of the facility with respect to its regulations and regulatory guidance, license and licensing process, and inspection program. These deficiencies contributed to the NRC failure to prevent or identify the licensee problems before the May 29 event. The NRC reported that it was reviewing the findings of the investigation "to develop an appropriate set of lessons learned both within NRC, the industry and the licensee."

ACNW COMMENTS ON SEVERAL ISSUES

The Advisory Committee on Nuclear Waste (ACNW) sent three letter reports to the NRC during July, August, and September 1991. One of these will be briefly discussed and excerpted here. The other two provide a program plan of anticipated ACNW activities for September–December 1991 (Ref. 16) and comments regarding 10 CFR Part 61 Proposed Revisions Related to Groundwater Protection.[17]

ACNW Comments on EPA Standards on Spent Fuel and HLW

The ACNW reviewed and provided comments on the six questions that accompanied Working Draft 3 of the proposed Environmental Protection Agency (EPA) Standards (40 CFR Part 191) for the management and disposal of spent nuclear fuel and high-level and transuranic radioactive wastes. An excerpt from the letter reports embodying the ACNW comments on the six questions is as follows:[18]

Question 1
Two options are presented in Sections 191.03 and 191.14 pertaining to maximum exposures to individuals in the vicinity of waste management, storage and disposal facilities: a 25 millirems/year ede limit and a 10 millirems/year ede limit. Which is the more appropriate choice and why?

Response:
The question, as phrased, refers to "maximum" exposures to "individuals." Because radionuclide releases from a high-level waste (HLW) repository, if they occur, could continue for a number of years, we have responded to the question in the sense of what would be the maximum acceptable annual exposure (dose) to members of the public over an extended period of time, in contrast to what might be considered an acceptable maximum exposure over a single year. This is in accord with the approach taken by both the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP).

In a similar manner, we assume that by maximum exposures to "individuals," the EPA means maximum exposures to a "critical population group," following the approach recommended by the ICRP. With those caveats, our response follows.

We believe an effective dose rate limit of 0.10 mSv (10 mrem) per year is more appropriate for several reasons:

1. Recent evaluations indicate that the biological effects of ionizing radiation may be higher than previously estimated.

2. The population in question may be exposed to more than one radiation source.

3. A fraction of the current dose limit should be reserved for potential future radiation sources.

4. Radionuclide releases from a repository, if they occur, could continue over a long period.

Such a dose rate limit would also be consistent with the recommendations of international organizations such as the ICRP, the International Atomic Energy Agency, and as noted in the 1989 report prepared by the radiation protection and nuclear safety authorities of Denmark, Finland, Iceland, Norway and Sweden (commonly referred to as the "Nordic" Study).

Question 2:
A new assurance requirement is presented in Section 191.13 that would require a qualitative evaluation of expected releases from potential disposal systems over a 100,000-year timeframe. Are such evaluations likely to provide useful information in any future selecting of preferred disposal sites?

Response:
We recognize that the specification of the 10,000-year time limit is somewhat arbitrary. It is important that significant geologic or climatic changes do not occur in the near-term period following the 10,000-year limit. We also agree that many geologic and climatic events that may affect the evaluation of site performance can be meaningfully extended beyond 10,000 years. In these cases, such an extension could provide information that would be useful for comparing the relative merits of several potential repository sites. In general, however, and particularly in the evaluation of the merits of a single site, the uncertainties involved in such an extension would make the value of the associated assessments questionable. It is important to note that, although evaluations of site performance may be quantitative, the results are subject to interpretation.

Question 3:
Two options are presented in Section 191.14 and 191.23 pertaining to the length of time over which the individual and ground water protection requirements would apply: a

1,000-year duration and a 10,000-year duration. Which is the more appropriate timeframe and why?

Response:

Title 10 Part 60 of the NRC regulations specifies that containment of the radionuclides within the waste be substantially complete for a period not less than 300 years nor more than 1,000 years. This constraint, coupled with other requirements, including the stipulation that the groundwater travel time to the accessible environment be at least 1,000 years, is designed to ensure that protection of the individual and the groundwater will extend well beyond 1,000 years.

When one also considers the fact that, after only a few thousand years of decay, the health hazards of the high-level wastes will be no greater than that of the original unmined uranium ore, it becomes readily apparent that it should be possible to ensure individual and groundwater protection for a duration of 10,000 years. We therefore endorse the extension of this time period. Such an extension would also make this requirement compatible with the limitation on health effects resulting from an HLW repository.

Question 4:

In Subpart C the Agency proposes to prevent degradation of "underground sources of drinking water" beyond the concentrations found in 40 CFR Part 141—the National Primary Drinking Water Regulations. The Agency is aware, however, that there may be some types of ground waters that warrant additional protection because they are of unusually high value or are more susceptible to contamination. Should the Agency develop no-degradation requirements for especially valuable ground waters? If so, what types of ground waters warrant this extra level of protection?

Response:

We agree that pollution of "underground sources of drinking water" should not be permitted beyond the limits specified in the National Primary Drinking Water Regulations. We believe that a no-degradation requirement for certain large volume aquifers, that represent major long-term existing or potential drinking water sources, may represent undue stringency. A preferred approach would be to reject as potential sites for the storage or disposal of high-level radioactive wastes those land areas which, if contaminated, could have the potential for polluting such aquifers. However, the volume and present value of an aquifer should not be the sole criteria for identifying those that should be protected. Other criteria may become significant with the passage of time.

At the same time, we believe it is important to recognize that the dose rate from underground sources of drinking water, even if contaminated to the limits specified in the National Primary Drinking Water Regulations, would still contribute only a small fraction (4 percent) of the current long-term dose rate limit for members of the public. Even considering the more restrictive limit for an HLW repository (as suggested in our response to Question 1 above), groundwater complying with the Drinking Water Regulations would contribute no more than 40 percent of the dose rate limit. In this sense, application of the Drinking Water Regulations to a repository represents a degree of stringency, especially because the primary pathway for public exposures from such facilities is through drinking water.

Question 5:

Two options are presented in Notes *l(d) and (e) of Appendix B pertaining to the transuranic waste unit: a 1,000,000 curies option and a 3,000,000 curies option. Which is the more appropriate TRU waste unit and why?

Response:

The number of curies of transuranic waste that would be comparable to 1,000 MTHM of spent fuel ranges from 1 to 6 million curies, depending on when the assessment is made. Accordingly, we believe that it would be reasonable to adopt the 3 million curie option.

Question 6:

The Agency is investigating the impacts of gaseous radionuclide releases from radioactive waste disposal systems and whether, in light of these releases, changes to the Standards are appropriate. To assist us in this effort, we would appreciate any information pertaining to gaseous release source terms, chemical forms, rates, retardation factors, mitigation techniques and any other relevant technical information.

Response:

Two reports that may be helpful are

1. W. B. Light, et al., "C-14 Release and Transport from a Nuclear Waste Repository in an Unsaturated Medium," Lawrence Berkeley Laboratory, Report LBL-28923 (June 1990).

2. W. B. Light, et al., "Transport of Gaseous C-14 from a Repository in Unsaturated Rock," Lawrence Berkeley Laboratory, Report LBL-29744 (September 1990).

In commenting on this subject previously, we have noted the following:

a. The total inventory of carbon-14 in a repository containing 100,000 MTHM is estimated to be about 100,000 curies. This compares to a global production of carbon 14 by cosmic radiation of 28,000 curies per year, a global inventory of about 230 million curies, and an atmospheric inventory of 4 million curies. In fact, release of all of the carbon-14 inventory in a repository would increase the atmospheric inventory by only about 2 percent; this compares to natural variations in the atmospheric inventory of 10 percent to 40 percent.

b. Based on an assumed inventory of 100,000 MTHM, the rate of release of carbon-14 from a repository that would be permissible under the existing EPA Standards would be about 1 curie per year. Experience shows that any carbon-14 that is released would rapidly mix in the atmosphere, and estimates are that the accompanying dose rate to a person on top of Yucca Mountain would be far less than 0.01 mSv (1 mrem) per year. We also note that the limit on the release rate of 1 curie per year for a repository compares to an average release rate of 10 curies per year from a typical 1,000 MWe light-water reactor.

At the time the EPA Standards were developed, considerations were limited to evaluations of a saturated site. In such a case, water transport and geochemical barriers would have been strongly influential in retaining the carbon-14. Subsequent consideration of Yucca Mountain (an unsaturated site) makes the existing EPA Standards inappropriate. We believe the limit for carbon-14 as specified in the proposed Standards should be relaxed. For additional discussion on this topic, we refer you to the transcript and minutes of the Advisory Committee on Nuclear Waste Working Group meeting held on March 19, 1991.

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Operating Experiences

Edited by G. A. Murphy

Aging Assessment of BWR Control Rod Drive Systems[a]

By R. H. Greene[b]

Abstract: *This Phase I study for the Nuclear Plant Aging Research (NPAR) Program examines the aging phenomena associated with boiling-water-reactor (BWR) control rod drive mechanisms (CRDMs) and assesses the merits of various methods of managing this aging. Information for this study was acquired from (1) the results of a special CRDM aging questionnaire distributed to each U.S. BWR utility, (2) a first-of-its-kind workshop held to discuss CRDM aging and maintenance concerns, (3) an analysis of Nuclear Plant Reliability Data System (NPRDS) failure cases attributed to the CRD system, and (4) personal information exchange with industry experts.*

An eight-page questionnaire was prepared by the Oak Ridge National Laboratory (ORNL) and distributed to all domestic BWR plants. The survey solicited site-specific data on CRDM degradation and failure experience, maintenance and aging interactions, and current testing procedures. For first-hand information on CRDM aging histories, a workshop was sponsored by ORNL to discuss CRDM performance and the overall questionnaire results with utility participants. The 3-day meeting on CRDM aging and maintenance was attended by 26 utility personnel from 21 BWR plants and 14 vendor and commercial representatives. These attendees provided invaluable information needed for understanding degradation mechanisms and maintenance constraints associated with BWR CRDMs.

As part of this study, nearly 3500 NPRDS failure reports have been analyzed to examine the prevailing failure trends for CRD system components. An investigation was conducted to summarize the occurrence frequency of these component failures, discovery methods, reported failure causes, their respective symptoms, and actions taken by utilities to restore component and system service.

The results of this research have identified the predominant CRDM failure modes and causes. In addition, recommendations are presented that identify specific actions utilities can implement to mitigate CRDM aging. An evaluation has also been made of certain maintenance practices and tooling that have enabled some utilities to reduce as low as reasonably achievable (ALARA) exposures received from routine CRDM replacement and rebuilding activities.

Control rod drive mechanisms (CRDMs) are located at the bottom of boiling-water-reactor (BWR) pressure vessels, and they position the neutron-absorbing control rod assemblies (CRAs) within the reactor core to provide reactivity control during startup and shutdown of the reactor, flux shaping at power, and emergency shutdown (scram). The control rod drive (CRD) system consists of the CRDMs; the hydraulic control units (HCUs); and various valves, pumps, and headers that supply, move, and retain the system's operating fluid.

The CRDM is a double-acting, mechanically latched, hydraulic cylinder that uses reactor quality water as its operating medium. Each CRDM has a companion HCU that contains numerous valves to regulate the operating flows and pressures delivered to the device. A CRA is attached to each CRDM at the spud, and movement is

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accomplished by admitting pressurized water into the appropriate part of the CRDM (Fig. 1).[1] The drive mechanism is capable of inserting or withdrawing a CRA at a slow, controlled rate to vary reactor power, or it can provide scram insertion to accomplish rapid shutdown of the reactor within a few seconds.

General Electric has manufactured six different models of CRDMs and four basic models of HCUs that are in service at BWRs throughout the United States. Improved scram times and enhanced operational performance have been the bases for many of the design differences occurring among the various models of both the CRDM and the HCU. Some aging-related degradation reported in the BWR-2, -3, -4, and -5 design CRDMs has been substantially reduced by material improvements and design features inherent to the BWR-6 models. Other types of reported component degradation are subject to plant operational parameters, such as water chemistry, and vary in frequency of occurrence with each BWR unit.

Normal CRDM maintenance involves the overall cleaning and replacement of a relatively standard set of components with new or spare parts. If necessary, any part of the CRDM can be replaced during rebuilding activities. Several utilities have established maintenance goals that require the refurbishment of all the CRDMs in a BWR unit every 10 years. However, historical data suggest that the maintenance interval varies for CRDMs with respect to their location in the core: centrally located drives are rebuilt more often than drives located along the periphery. The cause for dissimilar maintenance intervals is uncertain, but, as the result of lower vessel head geometry, the centrally located drives have more surface area

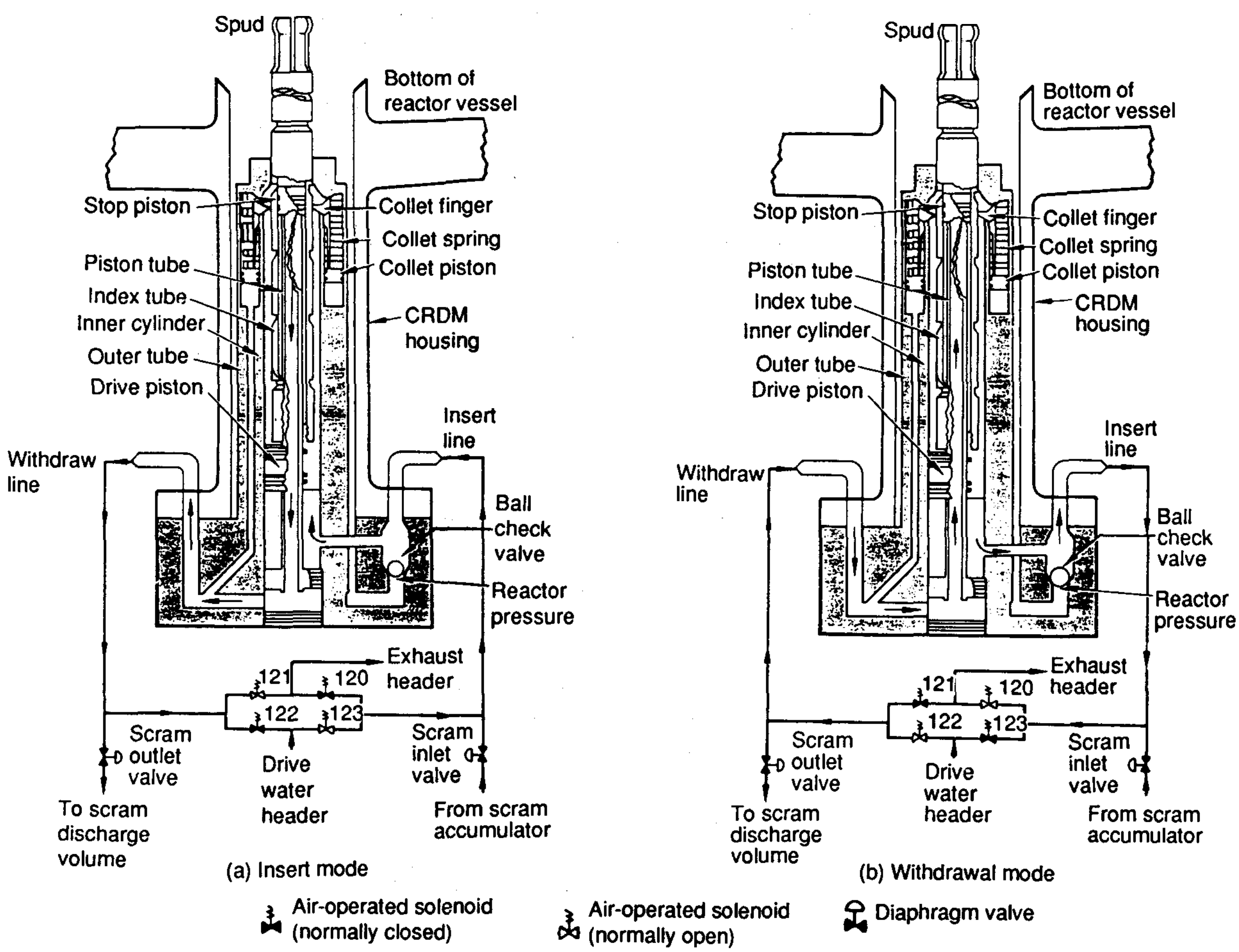

Fig. 1 Control rod drive mechanism operation.

exposed to the inside of the reactor than peripheral drives do and may experience higher temperatures along this exposed length.

Selection criteria for CRDM changeout does vary between plants. For the monitoring of service wear and degradation, most utilities routinely trend individual CRDM withdrawal stall flows and operating temperatures. In addition, plant Technical Specifications require scheduled scram-time testing and weekly-to-monthly CRDM "exercise" tests to ensure operability. In general, CRDM components degrade slowly as they age, and most aging problems do not occur suddenly but over a time interval of several years.

When a CRDM's performance indicators (e.g., stall flows, operating temperatures, and scram timing) begin to decline, it is scheduled for maintenance, usually during the next plant refueling outage. In recent years advancements in maintenance tooling, in changeout and rebuilding training, and in CRDM handling devices and improvements in worker comfort have significantly decreased the human error contribution to CRDM aging as well as reduced as low as reasonably achievable (ALARA) exposures. The following sections highlight the predominant modes of CRD system degradation and specific steps taken by utilities to mitigate component aging and curtail maintenance-related doses. Final research results will be published in NUREG/CR-5699, Vol. 1, entitled *Aging and Service Wear of Control Rod Drive Mechanisms for BWR Nuclear Plants.*

CRDM DEGRADATION: CAUSES AND CORRECTIVE ACTIONS

As a whole, the 21 nuclear plants that responded to the CRDM aging workshop questionnaire reported a good performance history for the BWR CRDMs. Nuclear Plant Reliability Data System (NPRDS) data analysis confirms this observation, with 72% of the failure reports being discovered by scheduled testing or routine observation, 24% by control room personnel, and only 2% as a result of a failed-service demand (the remaining 2% of the NPRDS reports did not identify a discovery method in the failure narrative). The term "failure" applied in these NPRDS case histories refers to a component malfunction that may range in consequence from relatively insignificant (e.g., a small valve stem leak) to a complete operational failure (e.g., the valve did not perform on demand).

Workshop participants were also asked to share observations regarding the primary causes of CRDM aging. In addition to normal service wear, the reported causes of CRDM degradation are Graphitar seal embrittlement, fatigue fracture, thermal degradation, collet housing cracking, nitrided surface corrosion, human errors made during drive changeout and rebuilding activities, and, to a lesser extent, plastic deformation caused by improper storage methods.

Debris, Corrosion, IGSCC, and Water Chemistry

Primary system cooling water containing "crud" (dirt particles, debris, corrosion products, and foreign materials that are found in varying amounts in the coolant) is ingested into the CRDM and is responsible for the degradation of several components. Corrosion usually occurs first on CRDM components with nitrided surfaces: the index tube, piston tube, guide cap, and collet assembly. Debris becomes entrapped in the CRDM during normal operations, and its presence scars metal surfaces and defaces the Graphitar seals.

As crud accumulates in the CRDM, the device's coolant flow rate may decrease and cause drive temperatures to increase, which contributes to the thermal degradation of the seals. After a scram, coolant flow rates may increase and CRDM temperatures decrease because some of the crud has been "shaken out" of the drive. In addition, entrapped crud between the Graphitar seal sets and their seating surfaces (on the drive and stop pistons) creates uneven force distributions during scram impacts that can cause seals to improperly function and break.

Some utilities are vacuuming the bottom of the reactor vessel inside the guide tubes during refueling operations to reduce the amount of crud that can become entrapped in the CRDM. In addition, the pre-BWR-6 design CRDMs had problems with the cooling water orifices becoming plugged with crud, which caused increased operating temperatures. Many utilities have retrofitted the older CRDMs with upgrade kits that modified the design of the cooling water orifice to mitigate this potential problem.

Nitrided surface corrosion has also been aggravated by poor storage methods. Occasionally, CRDMs are stored wet in air for more than 30 days before they can be rebuilt, usually as the result of strained outage schedules. The nitrided surfaces of the CRDMs begin to corrode, and the drive becomes excessively hard to disassemble for rebuilding. CRDM components can be inadvertently damaged by mishandling during a difficult disassembly process. One utility is currently using a long-term storage technique that places its "dirty" CRDMs in an aqueous solution of triethanolamine, a corrosion inhibitor, so that rebuilding can be delayed for up to 24 months. Some advantages for using this type of storage technique are: (1) It allows for radioactive decay, (2) it permits maintenance to be performed off the critical path, and (3) it enables drives to be rebuilt within 30 days of their actual need. According to workshop participants, this method of storage has been used several times before with no noticeable component deterioration.

Significant numbers of CRDMs have been retired from service because of collet housing cracking. This intergranular stress corrosion cracking (IGSCC) phenomenon has been found in model A, B, and C drives that were originally installed in BWR-2, -3, -4, and -5 plants. In the questionnaire, one utility reported that 46% of the cylinder, tube, and flange assemblies in its CRDMs had to be replaced because of this type of degradation. Later CRDM designs (models D, E, and F) that were supplied in BWR-6 plants changed this component's material from a 304 to a CF3 (cast 304L) stainless steel. These improved models have not experienced this problem. Utilities observing collet housing cracking in their drives have either replaced affected CRDMs with the later model drives or improved the earlier models with upgrade kits from the vendor.

In the past decade, water chemistry in the primary system has been modified by hydrogen injection in at least nine BWR facilities. This practice is intended to reduce the potential for primary system corrosion and IGSCC but was not implemented to address problems with CRDM collet housing cracking. For the reduction of the probability of IGSCC of the CRDM collet housing, the CRD system should use high-purity deaerated water (characterized by lower oxygen content and conductivity) during reactor operations; this is normally available from the condensate treatment system instead of the condensate storage tank.

Effects of Fatigue and Mishandling

Fatigue and/or mishandling are suspected to be the causes of certain effects observed in the spud, the CRDM component that engages the control rod assembly blade via the uncoupling rod. There have been reports of the "fingers" of this Inconel X-750 component being easily bent after a prolonged service history (> 15 years) in the reactor vessel. CRDM rebuilding technicians have described the effect as the fingers "losing their memory" and have used screwdrivers to pry and bend the fingers back into a proper concentricity (a practice that is not recommended). Although no professional metallurgical examinations have been conducted on a malformed spud, the cause of the bent fingers is speculated to be (1) fatigue caused by mechanical loads imposed by repeated scrams, (2) deformation resulting from mishandling during CRDM installation, or (3) deformation from CRA installation while the CRDM is partially inserted. This type of spud damage (as shown in Fig. 2) can present a myriad of coupling and uncoupling difficulties with the CRA. The spud, like all CRDM components, should be exchanged with a new spare part during rebuilding activities if it is damaged.

Nitrided Surface Degradation

In some CRDMs with a continuous service history greater than 15 years, degradation of the nitriding has been observed to the extent that, in one particular example, the unusually rough surface of an index tube could be easily scored with a piece of wood.[2] Although

Fig. 2 Bent control rod drive mechanism spud fingers (notice lack of concentricity).

no formal metallurgical investigations have been conducted to determine the nature of this effect, it could be the result of a combination of causes: prolonged radiation exposure, poor water chemistry, high operational temperatures, and variations in the case hardening from the nitriding process. No operational problems were reported for one CRDM with a longitudinally "striped" index tube, but the component was not reused in the rebuilt device because its continued serviceability was considered questionable.

Graphitar Seal Wear and Breakage

Replacement of the Graphitar seals is a standard requirement during CRDM rebuilding activities. An intact and correctly seated seal allows differential hydraulic pressures (upon withdrawal, insert, and scram signals) to position the drive. As these seals degrade and become less effective [i.e., become broken, scarred, or chipped as the result of numerous scram impacts; undergo normal surface wear; or experience thermal degradation caused by drive temperatures greater than 177°C (350°F)], the CRDMs stall flows increase, and greater hydraulic pressures are required to maneuver the drive. There were 275 NPRDS failure reports that cited Graphitar seal wear as the cause of deteriorating CRDM performance. The predominant location of the seal failures was on the stop piston. CRDM withdrawal stall flows over 316 $cm^3\ s^{-1}$ (5.0 gpm) (not attributable to the valving configuration on the HCU) are considered indicative of deteriorated seals that need to be replaced. Both General Electric and the Toshiba Corporation have developed improved Graphitar seals designed to be more durable and have a longer service life than those currently used in domestic BWR CRDMs. General Electric's new BWR-6 CRDMs are already equipped with these improved seals, and an improved replacement seal kit for BWR-2, -3, -4, -5, and -6 model CRDMs became available to utilities in February 1992.

Inner Filter Disengagement

Each CRDM has an inner and outer filter that serves to collect debris from reactor water that might otherwise damage the CRDM. The inner filter has been attributed with 90 failure reports in the NPRDS. Installation and maintenance errors were cited in 35 cases. Inner filters that are incorrectly installed during CRDM rebuilding can become loose during drive operation and cause the CRA to uncouple itself from the CRDM's spud. Uncoupling is a symptom observed in 27 failure reports.

The inner filter is mechanically attached to the stop piston by means of a spring clip (Fig. 3). When assembled, the inner filter engages the piston connector knob and is retained by locking flats that capture its spring clip after the filter is pushed onto the piston knob and rotated about 90° (Fig. 4). To test the proper installation of the inner filter, General Electric recommends using a filter assembly tool to pull the inner filter away from the stop piston with a force of about 89 to 133 N (20 to 30 lb). After engagement has been verified, the tool is removed from the CRDM, sometimes with an unintentional jiggling or twisting motion. When this is done, the filter becomes improperly oriented and can easily be disengaged. Even if the filter is not fully rotated 90°, the filter may inadvertently be rotated more during CRDM rebuilding and handling activities.

During the initial withdrawal venting of entrapped air for a reinstalled CRDM, the CRDM is inserted to a notch position less than 06 and then fully withdrawn back to position 48. If the inner filter was not truly engaged during the rebuilding process, it could bind against the inner surface of the CRDM index tube during CRDM withdrawal. In this scenario, the inner filter can become disconnected, cocked, and suspended. During the applied withdrawal signal, the uncoupling rod could jam against the side or top of the inner filter. When the CRDM is fully withdrawn at position 48, the misconfiguration of the internal components can result in the CRDM uncoupling with the CRA.

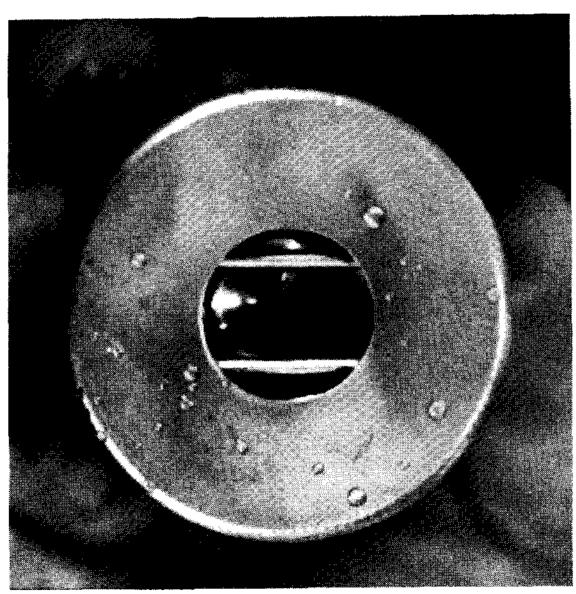

Fig. 3 Control rod drive mechanism inner filter spring clip.

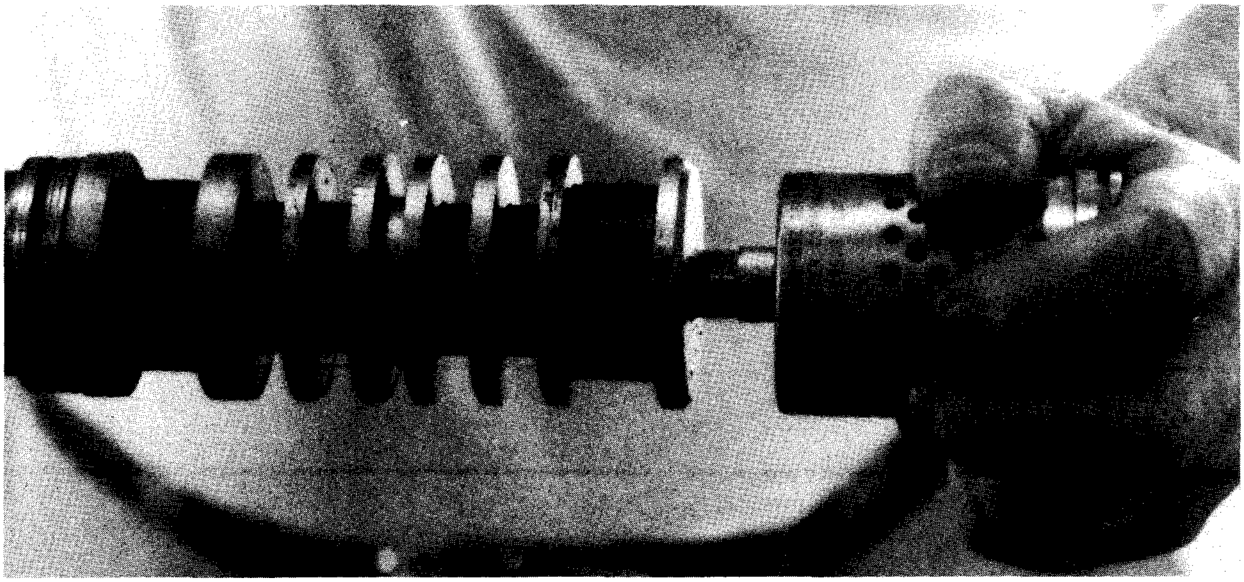

Fig. 4 Attachment of inner filter to stop piston.

The Toshiba Corporation has modified the base configuration of its inner filter to provide an improved design that would prevent uncoupling caused by misassembly. To date, there have been no design enhancements made in the attachment configuration of inner filters used in CRDMs operating in U.S. BWRs that would circumvent this type of disengagement.

Uncoupling Rod Misinstallation

Uncoupling has also been the result of the CRDM's uncoupling rod being misinstalled into one of the spud's flow holes instead of the center holes and thereby becoming vertically misaligned. This arrangement allows the bottom end of the uncoupling rod to contact the upper flange of the inner filter and thus lift the control rod lock plug sufficiently to cause uncoupling. An uncoupling rod incorrectly installed in this manner can also prevent the CRDM from being withdrawn to backseated position 48. If the uncoupling rod jams inside the spud flow hole from nonvertical orientation, it can stop the downward movement of the index tube before it reaches position 48. General Electric introduced a new uncoupling rod design in 1989 that was developed to prevent incorrect installation. The improved rod is available for BWR-2 to BWR-6 model CRDMs (Ref. 3).

Improper CRDM Storage Methods

The 4.7-m, 204-kg (15.5-ft, 450-lb) CRDM can be a challenge to store. Inadequate storage support has been blamed for a few observed cases of CRDM "sagging." These cases were confirmed by performing runout measurements along the length of the drive. Utilities store CRDMs in shielded vaults, on specially built racks, and sometimes in their original shipping crates. CRDM components can be damaged by laying drives on the floor with only the collet housing and the flange end supporting the weight, as shown in Fig. 5. CRDMs should not be stacked on top of each other separated by wooden blocks; this, in essence, transmits the weight of the stack to the lowest drive. Heavy, lead shielding "pigs" are sometimes left hanging on the spud end of "hot" drives for long periods of time; this places a moment on the collet housing. As shown in Fig. 6, CRDMs should be stored in racks or vaults with a minimum of 2 points of support located 61 cm (24 in.) from the flange end and 137 cm (54 in.) from the spud end.

HCU DEGRADATION: CAUSES AND CORRECTIVE ACTIONS

The NPRDS analysis yielded specific information on HCU degradation. Over 59% of the CRD system failures are attributable to the HCU. The HCU components requiring the most maintenance and replacement (as reported in the NPRDS) are identified in Fig 7. The following information discusses the HCU components requiring the most maintenance and the causes of their aging.

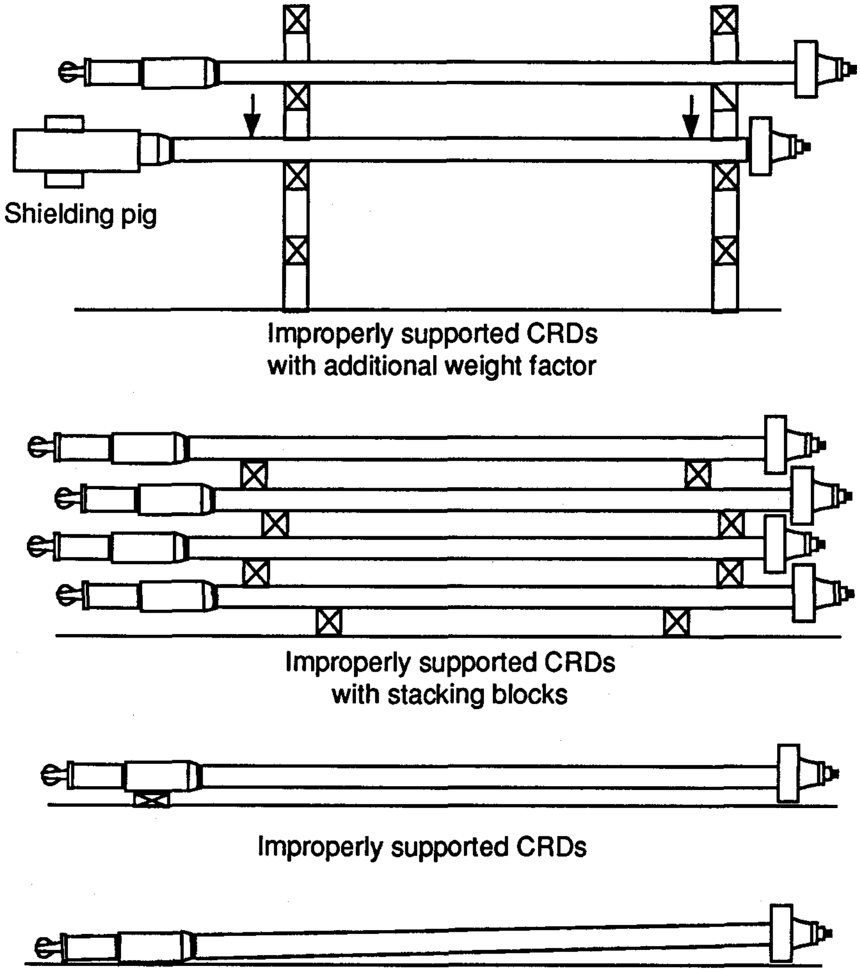

Fig. 5 Improper control rod drive mechanism storage scenarios. CRD, control rod drive.[2]

Accumulator Nitrogen Charging Cartridge Valve (HCU Part No. 111)

There were 526 NPRDS failure reports on the accumulator nitrogen charging cartridge valve. The leading reported cause of failure was attributed to worn valve packing (189 cases—36%). Normal valve wear or aging was second in the cause category (164 cases—31%), and a worn valve stem ranked third among failure causes (71 cases—14%). Additional reported failure causes were multiple-cause valve aging (cites the failures of several valve parts), valve seat aging, and worn valve seals.

The cartridge valve is located at the bottom of the HCU on the instrumentation block. This component is frequently referred to as the "star valve" because of the shape of the hand crank on the stem. Many of the failures of the "U-cup" packing may be attributed to incorrect installation. General Electric manufactures a four-part packing installation tool that was specifically designed to replace the U-cup packing in this valve. If the packing tool is not used when repacking the valve, it is easy to damage the packing on the valve stem threads during installation; this creates a new leak. It has also been reported that utility maintenance personnel occasionally adjust the star valve with their feet rather than bending over and using their hands. This practice could easily bend the narrow valve stem in addition to damaging the packing.

Scram Water Accumulator (HCU Part No. 125)

The NPRDS has recorded 189 failure reports of the scram water accumulator, with 119 of them requiring replacement units. In the pre-BWR-6 models, the chromium-plating liner of this carbon steel tank is porous enough to allow water to seep in and cause corrosion of

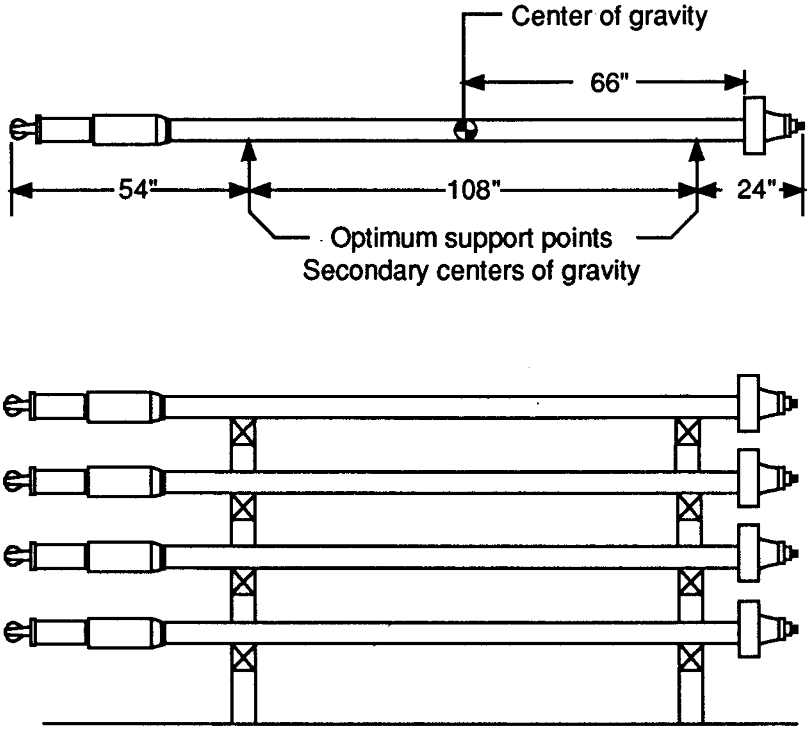

Fig. 6 Acceptable control rod drive mechanism storage arrangement.[2]

the carbon steel. General Electric issued a service information letter regarding the interior surfaces of these accumulators and determined that high-chloride, low-pH water conditions would produce blistering and pitting of the plating throughout the cylinder. It was further reported that loose flakes of this plating may leave the accumulator and collect on the Teflon seat of the inlet scram valve and cause some leakage. If this occurs, it can result in control rod insertion. In addition, the tank's corrosion flakes can etch Teflon from the scram valve seat and subsequently become entrapped in the cooling water orifice of the companion CRDM. General Electric and the Toshiba Corporation have developed stainless steel replacement units for this component. The predominant symptom of accumulator degradation reported in the NPRDS is a high-water-level alarm for the accumulator.

Inlet and Outlet Scram Valves (HCU Part Nos. 126 and 127)

There were 129 failure reports on the inlet scram valve (No. 126). The primary causes of degradation identified in the NPRDS were aging of the valve seat, multiple valve parts aging, worn valve packing, and worn valve diaphragms. Almost 65% of these reported failures have required valve rebuilding or replacement. As previously discussed, flakes of plating from a corroded accumulator can collect and erode the Teflon seat of this valve. In addition, the diaphragms of this valve are made from Buna-N reinforced with nylon. In a service information letter issued on this valve, General Electric recommended the lifetime (elapsed time between diaphragm cure and installation plus time in service) of this component to be 15 years for BWR-2s to BWR-5s and 12 years for BWR-6s. A supplemental service information letter from General Electric also stated that the nylon fibers around the diaphragm center hole on the Hammel–Dahl scram valve diaphragms could be damaged by the valve stem thread during diaphragm installation if the stem nut is tightened with the spring force applied under the diaphragm button. A subsequent redesign of the diaphragm has eliminated the protrusion of nylon fibers from the center hole.

The outlet scram valve (No. 127) had 77 failure reports that cited incorrect operation, worn seats, worn diaphragms, and worn stems and packing. More than 85% of the outlet scram valve failures reported in the NPRDS have required rebuilding or replacement to restore service.

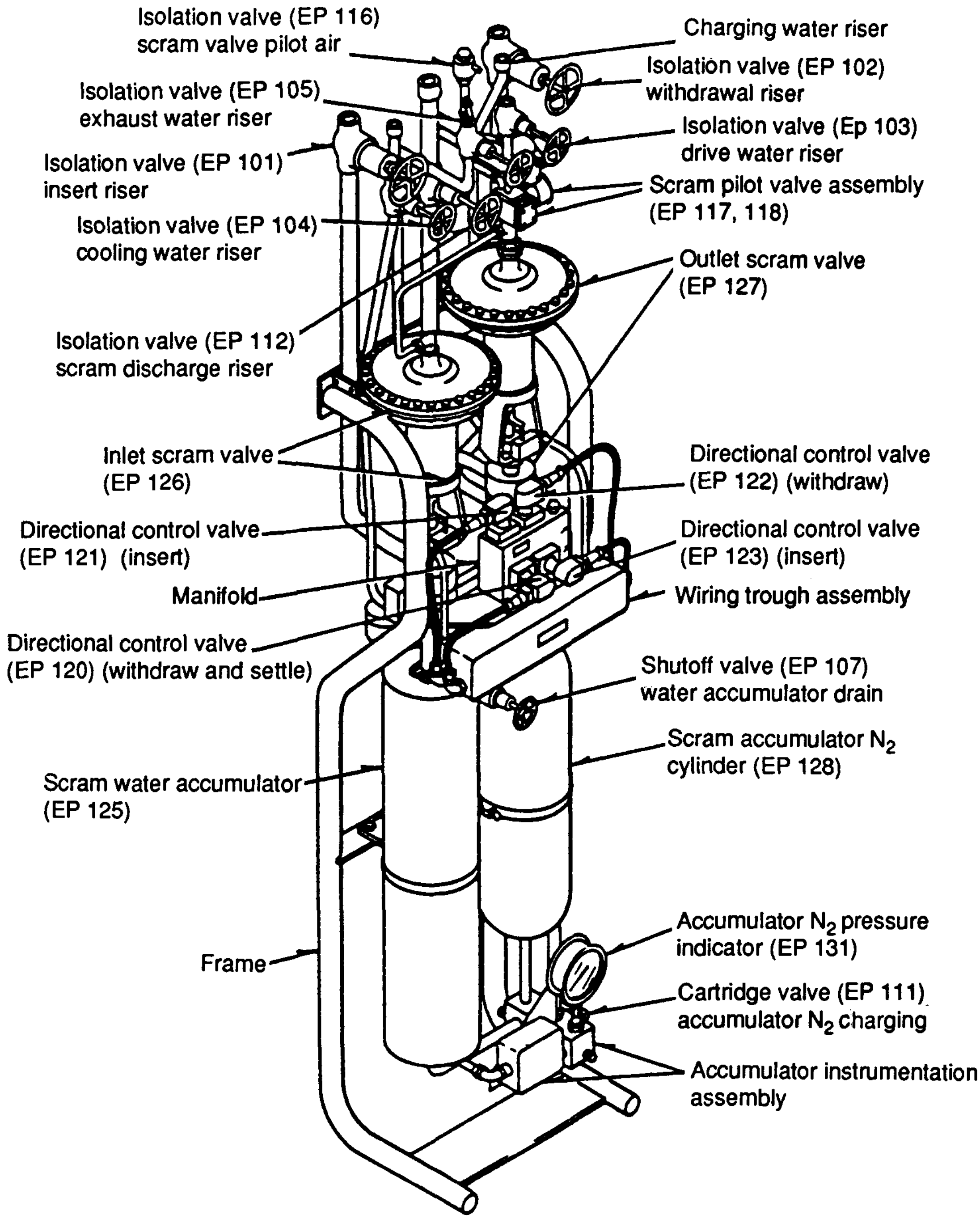

Fig. 7 Boiling-water-reactor hydraulic control unit (HCU) and its components. EP, equipment part.[4]

Scram Pilot Valve Assemblies and Solenoids (HCU Part Nos. 117 and 118)

There were 71 and 69 failure reports on the Nos. 117 and 118 valves, respectively. The causes of failure observed most frequently for these valves were a worn diaphragm, aged solenoid components (such as a coil "short" or a "blown" fuse), and normal valve wear or aging. The scram pilot valve solenoids had 241 reports of failure (185 by one plant) that cited the primary causes of failure as a worn seat (or disk). General Electric issued a service information letter indicating that cracking of the Buna-N rubber disks had been observed at a BWR plant which caused delays in CRDM scram times. The cracking and deterioration of the Buna-N disk material were accelerated by long-term exposure to the heat of the normally energized solenoid coil and by oil and water contaminants in the instrument air supply of the utility.[5] Because there is a continuous heat source from these normally energized solenoids and because increasing temperatures can indicate imminent coil failure,[6] utilities could periodically monitor and determine the trend of surface temperatures to detect coil degradation, which was cited as the secondary cause of failure for these valves. Industrial pyrometers could be used to obtain these data. General Electric also recommended in a service information letter that all BWR utilities establish a preventive maintenance program to replace all core assemblies, diaphragms, and associated parts in all CRD scram pilot valves, backup scram valves, and scram discharge volume test valves at periodic intervals because the Buna-N parts in these valves have a combined 7-year shelf and in-service life that elapses from the packaging date on the rebuild kit. The symptoms of scram pilot valve and solenoid degradation include slow scram times, leaking air, and abnormal solenoid noise (chattering, rattling, or a-c hum).

BALANCE-OF-CRD-SYSTEM COMPONENT FAILURES

If a failure report cited in the NPRDS were not attributed to a component associated with either the HCU or the CRDM, this analysis effort classified it as a balance-of-CRD system (BOCRDS) component failure. Only 18% of the failures reported in the NPRDS were attributed to this category. The following paragraphs discuss the categories that had the highest numbers of failure reports.

CRD System Pumps and Pump Components

The CRD system pumps and pump components had 117 failure reports in the NPRDS. Worn bearings, seals, piping and parts erosion, looseness, and normal wear or aging are the most prevalent problems identified in this data base. Over 98% of the pump failures were discovered by testing or routine maintenance. Several U.S. utilities (both BWR and PWR) have instituted monthly-to-quarterly vibration signature analysis programs on various types of rotating machinery in their stations as part of their overall maintenance and ALARA reduction efforts. Bearing anomalies, misalignment, unbalance, looseness, and soft foundations are readily analyzed and diagnosed with fast Fourier transform (FFT) analysis. Other programs augment their diagnostics by using oil analysis to examine degradation of metallic parts. Although there have been a few CRD pumps completely changed out, pump components have normally been replaced on an "as-needed basis" to restore service; occasionally, they have been entirely rebuilt.

Miscellaneous Scram Discharge Volume Valves

There were 44 failure reports on valves associated with the scram discharge volume. In 25% of these cases, the valve actuator or operator was simply out of adjustment. Over 27% of the reports cited entrapped debris as causing component failure. One station reported corrosion and entrapped debris on the scram discharge volume vent valve caused by a failure in procedures to regularly cycle the valve. For the alleviation of buildup of debris, procedures were enhanced to require quarterly timing and results trending of valve actuation. Another station reported a failure of the scram discharge volume drain valve caused by an accumulation of dirt and corrosion on the seat surface. The failure narrative reported that the maintenance staff felt this may have been caused by a prolonged shutdown. Scram discharge volume system component failures have also been attributed to contaminated instrument air used to operate system solenoid valves.[5] The majority of the scram discharge volume valve failures (80%) have required valve rebuilding or replacement.

HCU and BOCRDS Electrical Components

This section combines the results of the electrical component failures for the HCU and CRD system, including any electrical components associated with the CRD pumps. The group includes the reported failures of electrical relays, switches, controllers, transmitters, power supplies, circuit breakers, and fuses. There were a total of 207 failures—65 failure reports attributed to the HCU and 142 associated with the BOCRDS. There were no electrical component failure reports on the CRDM. The predominant causes of failure in these areas were electrical component aging or the device being out of calibration (includes setpoint drift). As might be expected, the component was restored to service either via adjustment or complete replacement.

CRD System Instrumentation

There were 79 reports of failed gages and instrumentation in the entire CRD system. As in the case of electrical components, the predominant causes of failure identified in the NPRDS were electronic component aging and out-of-calibration failures. Over 91% of these failures were corrected by an adjustment, and the remainder required a like-for-like replacement.

Selection Criteria for CRDM Changeout and Rebuilding

There is much debate regarding the criteria applied by utilities to select CRDMs for changeout and rebuilding. Although there are many contributing factors that may vary the rate and effects of CRDM aging, the recommended maintenance interval for a CRDM historically has been 10 years. With this figure in mind, many utilities have designed CRDM changeout schedules to reflect a 100% rebuild of all CRDMs in the reactor every 10 years. Other utilities, which are rigorously and routinely monitoring and trending stall flow rates and operating temperatures and acquiring friction traces of their CRDMs, believe they can confidently assess the operability of their CRDMs without scheduling drives for maintenance on the basis solely of elapsed service time. The workshop reviewed the CRDM changeout history for 20 BWR units. The data suggest that centrally located drives undergo more maintenance than peripheral drives. Attendees at the workshop stated that not all the CRDMs that had been changed out exhibited operational problems; frequently, operational problems that should have been directed at components on the companion HCU had been erroneously attributed to CRDMs.

The selection of CRDMs to be rebuilt can be initiated by classifying drives into two groups: *Priority 1 CRDMs*—those drives which *must* be exchanged or rebuilt, and *Priority 2 CRDMs*—those drives which *should* be exchanged or rebuilt and incorporated into the outage schedule if possible. Attendees at the workshop agreed on the following operational characteristics that would place suspect CRDMs into these two categories (see Tables 1 and 2).

Table 1 Characteristics of Priority 1 CRDMs—Must be Exchanged or Rebuilt

1. Excessive scram times—violation of plant Technical Specifications.
2. CRDM does not fully insert during a scram.
3. CRDM has a history of uncoupling.
4. CRDM will not go into position 48 (fully withdrawn).
5. CRDM consistently has a withdrawal stall flow greater than 316 cm^3 s^{-1} (5.0 gpm).

Table 2 Characteristics of Priority 2 CRDMs—Should be Exchanged or Rebuilt

1. Consistently high temperatures throughout length of travel [>177°C (350°F)].
2. Unacceptable withdrawal or insertion times that are unrelated to the HCU.
3. Repeated episodes of "double-notching" when moving, or CRDMs that continually require increased drive pressures to move (unrelated to the HCU).
4. CRDMs with high or abnormal friction traces not attributable to misalignment with fuel assemblies.

Workshop participants also stated that, when CRDMs began to display operational problems, several of the anomalies listed in Tables 1 and 2 would usually be manifested concurrently. For that reason, many utilities choose to rebuild CRDMs if they display *any* of the operational characteristics mentioned in these categories and might also include those drives which have a continuous service time of 10 years. Most CRDM aging problems have a long lead time and do not suddenly occur without exhibiting characteristic warning signals.

ALARA REDUCTION DURING CRDM CHANGEOUT AND REBUILDING

Workshop attendees commented that CRDM changeout and rebuilding is one of the highest-dose, most physically demanding, and most complicated maintenance activities routinely accomplished by BWR utilities. In the 30 years since the BWR design concept for commercial nuclear power production was first successfully demonstrated, there have been many improvements in the maintenance techniques used to pull and refurbish CRDMs. Some utilities, however, have not taken advantage of new tooling and continue to use outdated maintenance equipment that still adequately performs the task but inevitably results in higher doses delivered to the nuclear worker. According to questionnaire responses and nuclear commercial services input, substantial ALARA reduction can be realized by focusing improvements in three key areas associated with CRDM maintenance work: CRDM handling and exchange tools, worker comfort and environment, and worker training.

CRDM Handling and Exchange Tools

Five companies are currently offering pneumatically or hydraulically operated devices that can be placed in existing BWR undervessel work platforms to assist CRDM personnel with changeout activities. They replace conventional, electrically driven winch systems supplied with the plants and require only two technicians for equipment operation. More than half the sites responding to the questionnaire stated that they had either purchased or contracted for the use of this type of device in their CRDM changeout work. They also verified that the device had significantly improved the performance of CRDM maintenance. Most sites further stated that this type of device had reduced job-related exposures; two plants reported overall exposure reductions of 38 and 56%.

CRDM Worker Comfort and Environment

The CRDM aging questionnaire asked utilities to indicate which conditions during CRDM changeout most influenced improper CRDM maintenance. High temperatures were recognized by 65% of those participants as having the biggest negative impact on worker performance. In addition, high radiation levels (creating, in some cases, a false sense of urgency in workers not accustomed to this type of work), extremely cramped working conditions (a person works "hunched over" for long periods of time during changeout operations), poor vision (obstructed from instrumentation cabling and hampered by insufficient lighting), and inadequate communication were prevalent conditions that further complicate an already complex task. Other job location factors contributing to mishandling errors were disorientation, remoteness, cumbersome protection clothing, and visual impairment during CRDM "rainshowers" [the normal 126 to 189 $cm^3 s^{-1}$ (2 to 3 gpm) leak of reactor water when drives are removed from the vessel].

Several utilities have invested much time and money into developing improved maintenance conditions for CRDM changeout work. Some plants have revised and streamlined procedures; others are testing new designs of radiation protection clothing and portable air conditioning apparatus, installing temporary lighting, and developing specialized tools for these tasks. The overall consensus of the workshop attendees was that any utility that sought ways to improve worker comfort during these activities would realize benefits not only in ALARA reduction but also in fewer maintenance errors.

CRDM Worker Training

The CRDM worker training, particularly with undervessel mockups, improves crew performance and helps expedite tight outage schedules. "Full-rad dress" rehearsals are particularly valuable in acquainting technicians with working under restrained conditions. Both the CRDM changeout and rebuild crews should receive specialized training to correctly perform these tasks. More than half the participants responding to the questionnaire either trained their own crews on mockup assemblies or employed contractors who had completed similar training. Many of the utilities provide 3 to 5 days of training to crews involved in changeout and rebuilding activities. In some cases, shortened refresher courses are provided to personnel with previous experience. Several utilities stated that the training also involved individual testing. All those utilities providing training or using specialized crews verified that these activities yielded improvements in job performance. Other benefits mentioned were reductions in radiation doses, increased worker safety, improved worker attitude, and fewer rebuild errors.

Other modifications made by utilities to reduce radiation exposures acquired during CRDM changeout and rebuilding activities include the following:

1. Inner and outer filters were discarded as waste rather than cleaned.
2. Shielded inner and outer filter removal tools were used.
3. Flush tanks were used during CRDM rebuilding activities.
4. Installed ALARA shielding achieved reductions at several sites that historically have "hot" drives.
5. Shielded storage racks and/or customized concrete vaults have been built into CRDM rebuilding rooms.
6. Remote cameras installed under the vessel and in the rebuild room have helped to better coordinate activities, save time, and reduce exposures.

CONCLUSIONS

As a whole, BWR control rod drive mechanisms have a good service record at U.S. nuclear plants. The BWR-6 design CRDMs have incorporated modifications that have eliminated problems experienced by the earlier models. The primary causes of CRDM aging are embrittlement, fatigue fracture and thermal degradation of the Graphitar seals, nitrided surface corrosion, mishandling and rebuilding errors occurring during CRDM maintenance, and, to a lesser extent, improper storage support. According to NPRDS failure reports, the majority of maintenance for the CRD system occurs on the HCU. The HCU components reporting the most failures are the scram water accumulator, the accumulator nitrogen charging cartridge valve, the inlet and outlet scram valves, and their scram pilot valve assemblies and solenoids.

The CRDM changeout and rebuilding activities occur at all BWR nuclear plants but with varying amounts of worker exposures and time expended on the removal and refurbishment of drives. Many utilities are seeking ways to improve their CRDM maintenance processes. Some plants are aggressively pursuing ways to reduce radiation exposures acquired during CRDM maintenance by installing state-of-the-art tooling, improving worker comfort, and increasing maintenance training.

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In addition to the preceding references, the General Electric Service Information Letters (GE-SILs) on the CRD system are an excellent source of information. Utilities can contact their General Electric Site Services Representative for an index of the GE-SILs issued on the BWR system of interest.

Reactor Shutdown Experience

Compiled by J. W. Cletcher[a]

This section presents a regular report of summary statistics relating to recent reactor shutdown experience. The information includes both numbers of events and rates of occurrence. It was compiled from data about operating events entered into the SCSS data system by the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory and covers the three-month period of July, August, and September 1991. Cumulative information, starting from May 1, 1984, is also shown. Updates on shutdown events included in earlier reports are excluded.

Table 1 lists information on shutdowns as a function of reactor power at the time of the shutdown for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). Only reactors in commercial operation at the start of the reporting period (July 1, 1991) are included. The second column for each reactor type shows the annualized shutdown rate for the reporting period. The third and fourth columns list cumulative data (numbers and rates) starting as of May 1, 1984.

Table 2 shows data on shutdowns by shutdown type: *Real Scrams* are events in which the reactor was scrammed for a valid cause; *Spurious Scrams* are events in which an instrument failure or other fault causes a scram not actually called for by existing reactor conditions; *Non-Scram Shutdowns* (frequently from operating power to hot standby) do not involve actuation of the scram system either manually or automatically. Only reactors in commercial operation are included. The second column for each type of reactor 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

Table 1 Reactor Shutdowns by Reactor Type and Percent Power at Shutdown[a]
(Period Covered is the Third Quarter of 1991)

	BWRs (37)				PWRs (75)			
Reactor power (P), %	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
0	8	0.86	599	2.34	5	0.26	389	0.77
$0 < P \leq 10$	1	0.11	111	0.43	0	0.00	146	0.29
$10 < P \leq 40$	3	0.32	137	0.53	4	0.21	283	0.56
$40 < P \leq 70$	2	0.21	119	0.46	1	0.05	149	0.30
$70 < P \leq 99$	4	0.43	295	1.15	9	0.48	432	0.86
$99 < P \leq 100$	19	2.04	344	1.34	20	1.06	898	1.78
Total	37	3.97	1605	6.27	39	2.06	2297	4.56

[a]Data include shutdowns for all reactors of the designated type while in commercial service during all or part of the period covered. The cumulative data are based on the experience while in commercial service since the starting date of Jan. 1, 1984, through the end of the reporting period; it includes the commercial service of reactors now permanently or indefinitely shut down.

[b]Based on cumulative BWR operating experience of 256.08 reactor years.

[c]Based on cumulative PWR operating experience of 503.46 reactor years.

[a]Oak Ridge National Laboratory.

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 (July 1, 1991, for this issue). The first line of this table gives the information for reactors licensed for full power but not yet in commercial operation on that date.

Table 2 Reactor Shutdowns by Reactor Type and Shutdown Type[a] (Period Covered is the Third Quarter of 1991)

	BWRs (37)				PWRs (75)			
Shutdown (SD) type	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
SDs required by Technical Specifications	2	0.21	201	0.78	8	0.42	346	0.69
Intentional or required manual reactor protection system actuations	1	0.11	128	0.50	2	0.11	263	0.52
Required automatic reactor protection system actuations	24	2.58	753	2.94	21	1.11	263	0.52
Unintentional or unrequired manual reactor protection system actuations	0	0.00	9	0.04	0	0.00	18	0.04
Unintentional or unrequired automatic reactor protection system actuations	10	1.07	514	2.01	8	0.42	390	0.77
Total	37	3.97	1605	6.27	39	2.06	2297	4.56

[a]Data include shutdowns for all reactors of the designated type while in commercial service during all or part of the period covered. The cumulative data are based on the experience while in commercial service since the starting date of Jan. 1, 1984, through the end of the reporting period; it includes the commercial service of reactors now permanently or indefinitely shut down.

[b]Based on cumulative BWR operating experience of 256.08 reactor years.

[c]Based on cumulative PWR operating experience of 503.46 reactor years.

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age[a] (Period Covered is the Third Quarter of 1991)

	BWRs (37)						PWRs (75)					
Years in commercial operation (C.O.)	Exposure during the period (in reactor years)	Number: Reactors	Number: Shutdowns	Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year	Exposure during the period (in reactor years)	Number: Reactors	Number: Shutdowns	Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year
Not in C.O.[b]	0.252	1	0	0.00	330	28.13	0.000	0	0	0.00	334	35.21
First year of C.O.	0.000	0	0	0.00	121	9.00	0.252	2	2	7.94	276	10.15
Second through fourth year of C.O.	1.259	5	4	3.18	255	6.59	2.155	10	3	1.39	471	5.74
Fifth through seventh year of C.O.	1.763	7	7	3.97	112	4.78	3.387	14	8	2.36	246	3.87
Eighth through tenth year of C.O.	0.504	2	4	7.94	151	7.09	2.094	9	2	0.95	316	4.60
Eleventh through thirteenth year of C.O.	0.252	1	0	0.00	267	5.97	1.010	5	2	1.98	442	4.55
Fourteenth through sixteenth year of C.O.	1.081	5	7	6.46	378	6.42	3.097	13	10	3.23	319	3.49
Seventeenth year and over	4.406	18	15	3.36	321	5.77	6.897	28	12	1.74	223	5.12
Total	9.572		37	3.87	1935	7.23	18.891		39	2.06	2627	5.12

[a]Age is defined to be the time (in years) from the start of commercial operation to the time of the shutdown event, except for the first line, which lists reactors not yet in commercial service (see b below).

[b]This category includes reactors licensed for full-power operation but not yet commercial. During this reporting period reactors in this category included 1 BWR (Shoreham) and no PWRs.

Selected Safety-Related Events

Compiled by G. A. Murphy[a]

This occasional column in *Nuclear Safety* relates events with safety significance at U.S. nuclear facilities. In this issue two such events are included. One was a common-mode loss of instrument power at Nine Mile Point 2 on Aug. 13, 1991, and the other was a turbine over-speed event with turbine damage at Salem 2. Each event will be briefly described.

NINE MILE POINT 2 SUFFERS COMMON-MODE LOSS OF INSTRUMENT POWER[b]

Shortly before shift change on the morning of Aug. 13, 1991, an internal failure in the main transformer at Unit 2 of the Nine Mile Point Nuclear Power Plant[c] caused a turbine trip and reactor scram (i.e., automatic reactor shutdown). During the fraction of a second before automatic protective features isolated the transformer, the fault caused depressed voltages on the transmission system and on the in-plant electrical distribution system. Although of very short duration, the degraded voltage resulted in a simultaneous common-mode loss of five "uninterruptible" power supplies that powered important control room instrumentation and other plant equipment. Internal deficiencies, common to all five power supplies but unknown to the plant staff, had made the power supplies susceptible to failure initiated by degraded voltage. A simplified one-line diagram of the plant power distribution system is shown in Fig. 1. Nine Mile Point Unit 2 received a full-power operating license in July 1987. Unit 1, adjacent to Unit 2, is an older design with a separate control room, and its operation was not affected by the event.

Automatic reactor protection systems, including the scram, functioned properly. All necessary engineered safety features were available and used as needed. However, control-rod position indication was lost, and the operators took conservative action in accordance with their procedures as if there had been a failure to scram. The difficulty experienced by the operators because of the loss of many normally available plant status indications and equipment underscored the importance of the lost power supplies.

Within each uninterruptible power supply (UPS) that failed, control logic had functioned in response to the degraded voltage; this caused the UPS input and output power circuit breakers (CB-51 and CB-53) to open (Fig. 2). All the equipment powered from the five UPSs was consequently lost. The lost equipment included:

- All indications of reactor control-rod position, which resulted in the inability of the operators to verify that the reactor would remain shut down.
- Condensate and feedwater system controls, which resulted in main feedwater pump trips and loss of normal feedwater to the reactor.
- Virtually all control room alarm annunciators, which hampered the operators' ability to monitor post-scram operation of the plant.
- Both the in-plant radios and the page telephone communications systems, which limited control room communications with in-plant personnel.
- Control room indications of plant fire alarms, which required local monitoring of fire alarm panels.
- Almost all plant computers that perform monitoring, alarm, protection, and data recording functions, which reduced the operators' ability to monitor plant status, disabled some minor automatic functions, and made reconstruction of the event difficult.
- Multiple control systems, which resulted in a loss of normal containment space cooling and required that operators divert some attention to monitoring containment temperature.
- Many other parameter displays on the main control board, which limited the operators' ability to monitor plant conditions, particularly balance-of-plant (e.g., turbine; feedwater) equipment.
- The safety parameter display system, which removed an aid to operators for analyzing plant conditions and reduced information that was available in the Unit 2 technical support center.

[a]Oak Ridge National Laboratory.

[b]Extracted from NUREG-1455, *Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1991.*

[c]A 1080-MWe (net) GE BWR-5 operated by Niagara Mohawk Power Corporation located about eight miles east of Oswego, New York.

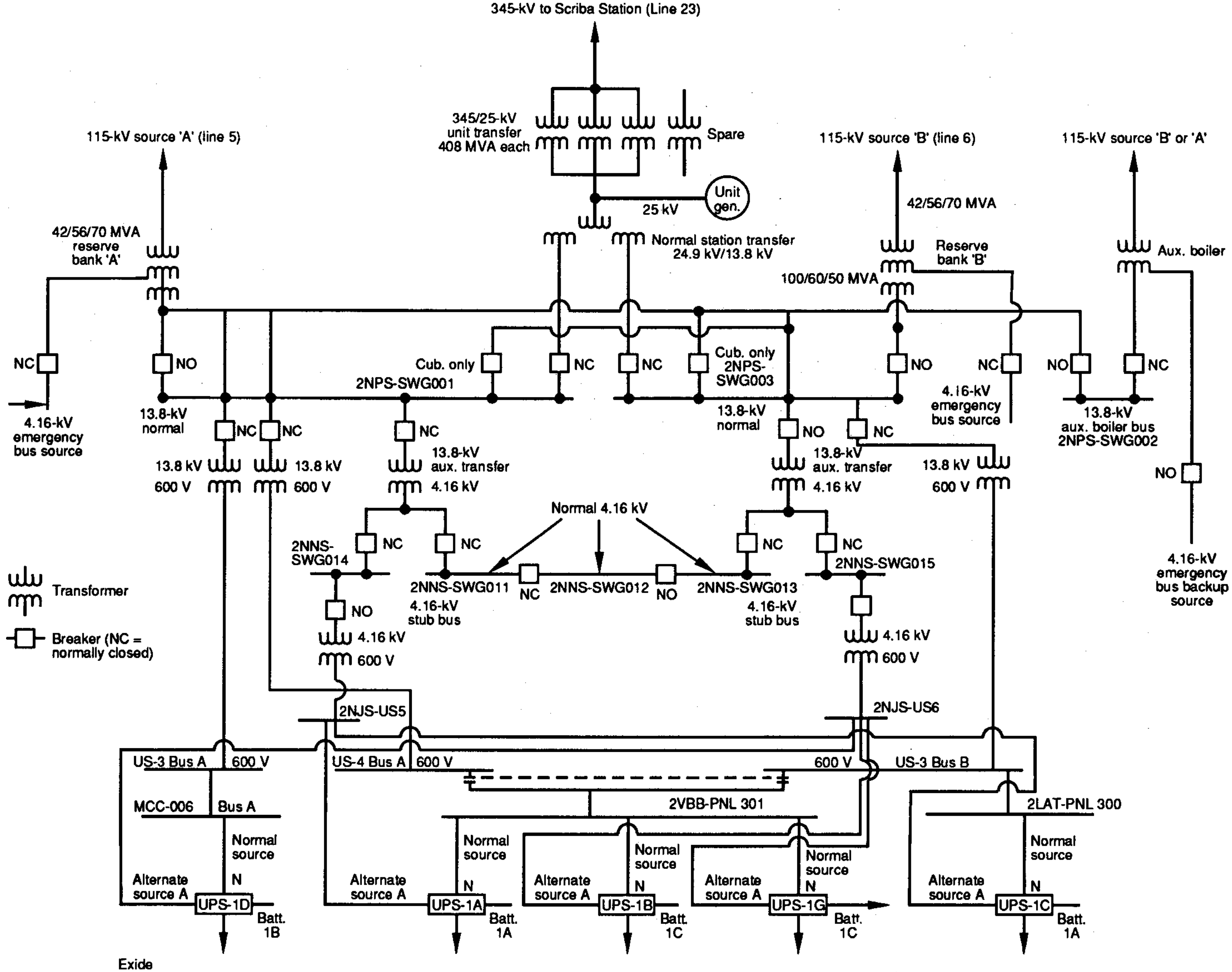

Fig. 1 Simplified one-line plant power distribution system diagram for Nine Mile Point 2.

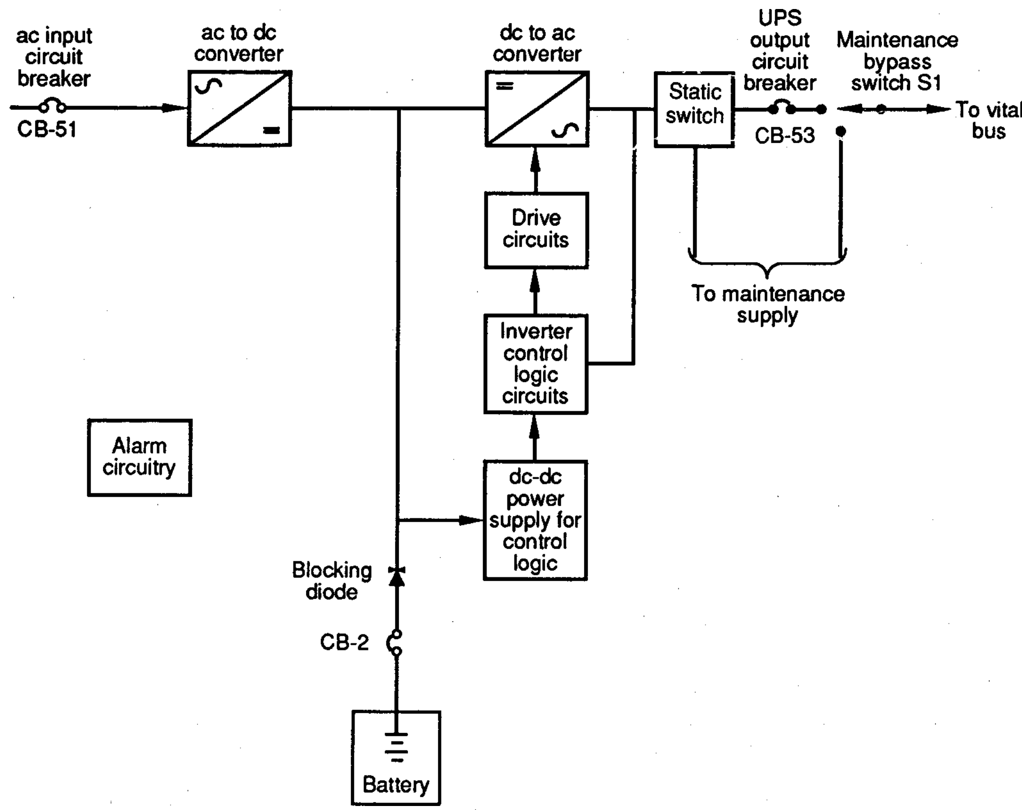

Fig. 2 Simplified block diagram for the safety-related 25-kVa uninterruptible power supply unit at Nine Mile Point 2.

• Some plant lighting that posed a personnel safety hazard but did not significantly affect plant personnel.

Control room operators responded to the loss of feedwater by starting the steam-turbine powered reactor core isolation cooling (RCIC) system pump and using applicable emergency operating procedures. The reactor began to depressurize as the result of the combined effects of cold water being sprayed into the reactor by the RCIC system and steam being drawn from the reactor by turbine building equipment and the RCIC pump turbine. Although the main feedwater pumps had tripped shortly after the start of the event, two condensate booster pumps remained operating. Valves remained open in the flow path from the condensate booster pumps through the idle feedwater pumps to the reactor. However, water did not immediately flow from this source because the reactor pressure was higher than the condensate booster pump discharge pressure.

As the reactor depressurized below the discharge pressure of the condensate booster pumps, a large uncontrolled volume of cold water was injected into the reactor vessel. Operators recognized the situation and stopped the condensate booster pumps before the injection had an adverse effect on the plant. In the very unlikely event that there had been an actual failure to completely scram the reactor (i.e., an anticipated transient without scram), however, the injection of cold water and accompanying positive reactivity addition could have resulted in significant consequences. When the cold water was injected, the operators did not know the position of any of the control rods because of the lost rod position indicating equipment, so they were proceeding as if a partial failure to scram had occurred.

The station shift supervisor (SSS) assumed the emergency director's responsibilities and declared a site-area emergency a few minutes after the event began. This emergency declaration was correctly based on a loss of important instrumentation, including annunciators, combined with a reactor transient. The notification of appropriate local, state, and federal emergency response organizations followed.

During the first minutes of the event, the control room operators were occupied with many tasks. The assistant SSS assumed the role of shift technical advisor. Within a few minutes the SSS was filling the roles of both control room supervisor and emergency director. The emergency director's responsibilities, lost instrumentation, efforts to restore electrical power, concern over control rod positions, and other demands placed a heavy burden on the SSS.

Meanwhile, control room operators diagnosed the cause of their lost instrumentation and other important equipment and dispatched field operators to inspect the UPS and to restore power. When they recognized that the UPS had tripped without transferring loads to alternate power, many of the operators at the UPSs did not know how to proceed. Procedures had not been developed to address the restoration of power following loss of a UPS, but at least one of the operators had sufficient understanding of the equipment to determine what was needed. Power was restored to the UPS loads a half hour after the event began, and operators subsequently verified that all control rods were completely inserted into the reactor core.

Following restoration of the UPS-supplied loads, the event proceeded as a relatively normal plant shutdown. Cold shutdown was achieved that evening at 6:46 p.m., and the site-area emergency was terminated at 7:43 p.m.

The Nuclear Regulatory Commission (NRC) initially dispatched a seven-member Augmented Inspection Team (AIT) to investigate the event. However, because of the apparent potential safety significance of the event, and to ensure that any generic technical and operational implications were understood, the Nuclear Regulatory Commission (NRC) Executive Director for Operations upgraded this activity to an Incident Investigation Team (IIT) on Aug. 15, 1991. The team was formed in conformance with the NRC Incident Investigation Program. The team, which included two industry representatives, was selected because of its broad experience in event analysis, with individual members having specific knowledge and experience in electrical power systems, including large transformers and uninterruptible power supplies, instrumentation and controls, boiling-water-reactor systems and operation, and human performance. The team was directed to determine what happened, identify the probable causes, and make appropriate findings and conclusions.

This event did not pose a threat to plant safety because the scram functioned properly to shut down the reactor. The significance of the event lies in the challenge that it presented to the operators the potential that severe challenges and resultant stress have to cause errors of omission or commission. The event is also significant because of the simultaneous failure and common-mode vulnerability of the multiple UPSs.

The team found that the event was caused and its course was shaped by several factors. Although the transformer failure was the initiating event, it should not be considered a cause in that transformer failure is an anticipated event for which nuclear power plants are designed to safely respond. However, the simultaneous loss of the five UPSs was unexpected and presented unique challenges to both equipment and personnel.

Two factors can be considered to be the direct causes of the UPS loss: (1) A design deficiency internal to each UPS and (2) failure of the plant staff to perform appropriate preventive maintenance. Within each UPS is a control logic unit that is essential to operation of the UPS units (Fig. 3). The UPSs were lost because the power for these control logic units was provided by a source that was affected by the degraded voltage resulting from the transformer failure. The control logic units can be supplied with backup power from internal batteries; however, these batteries were dead. Had either deficiency been corrected, the UPSs would not have been lost. All five UPS units are an identical design; hence all were vulnerable to a loss caused by degraded voltage.

The team examined operator performance and associated human factors. In this event the operators coped with a difficult situation and successfully addressed the problems they faced. They did, however, make some mistakes that were not safety significant because the reactor scrammed as designed. The operators should have prevented the injection of cold water from the condensate booster pumps. The team concluded that no one factor alone caused this problem; rather, the cause should be attributed to a combination of factors that acted synergistically to result in the operators' unawareness of the impending potential problem in time to prevent the injection. These factors included multiple demands for the operators' attention, the physical layout of the control board, procedure problems, and an unwanted reactor depressurization that was difficult to control. Also, corrective actions in response to previous similar uncontrolled condensate booster pump injections at this site have not been effective in preventing a recurrence of the problem.

In its investigation the team reviewed prior NRC activities and the licensee's response to regulatory communications that relate the loss of the UPS. The team found that the NRC had not presented a clear position to the regulated industry concerning control of equipment configuration and treatment of important balance-of-plant equipment nor had the agency performed an integrated

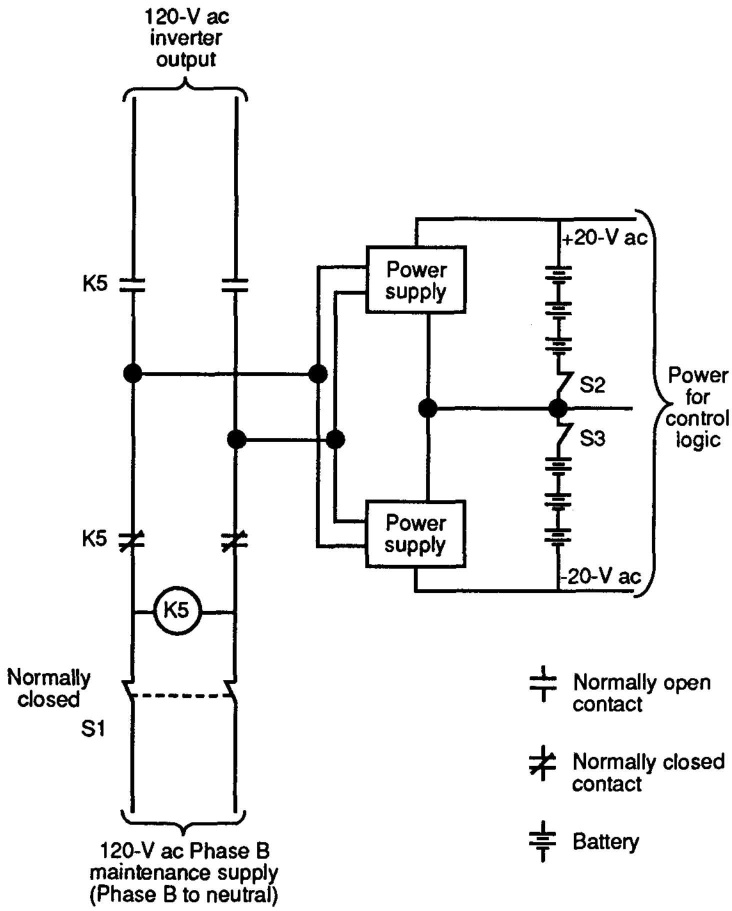

Fig. 3 Simplified diagram for the uninterruptible power supply control logic power supply at Nine Mile Point 2 at the time of the event.

review of instrument and control and operator actions. Such a review could have brought increased attention to the importance of control rod position indication and the challenges its inoperability could present to the operators.

SALEM 2 MAIN TURBINE DAMAGED[a]

On Nov. 9, 1991, Unit 2 of the Salem Nuclear Generating Station[b] was operating at 100% reactor power. At about 11:00 a.m., plant operators initiated a routine test procedure to verify the operability of the steam turbine automatic mechanical trip mechanisms. The test procedure involved the manipulation of mechanical trip devices in the turbine Auto Stop Oil (AST) system, the primary turbine protection mechanism. By design, the test procedure required the complete isolation of the AST system from any turbine control or trip function (including the mechanically actuated turbine overspeed trip device) to prevent an actual turbine trip during testing of the mechanical devices.

A redundant backup system for turbine overspeed protection and emergency trip functions was assumed to be operational. The backup system consists of three electrically actuated solenoid valves designed to provide redundant automatic control and trip of the turbine in an overspeed condition by reliance on two redundant overspeed protection solenoid valves (OPC-20-1 and OPC-20-2) and to cause a turbine trip on a reactor trip by reliance on a backup emergency trip solenoid valve (valve ET-20).

[a]Condensed from U.S. NRC Region I Report No. 50-311/91-81.

[b]An 1115-MWe Westinghouse PWR operated by Public Service Electric and Gas Co., located 8 miles southwest of Salem, New Jersey.

During the performance of the test, a momentary oil pressure perturbation (a pronounced decrease lasting about 1.5 seconds) occurred in the AST system. Though of short duration, the momentary oil pressure decrease was sufficient to open the AST Interface Valve. The AST Interface Valve functioned to relieve the Emergency Trip Fluid (ETF) pressure from the pilot valves affecting operation of turbine steam admission valves (i.e., Stop Valves, Governor Valves, Reheat Stop Valves, and Intercept Valves). Consequently those valves closed and isolated steam flow to the high- and low-pressure turbines.

The oil pressure perturbation also resulted in the activation of three low AST pressure signals to the Reactor Protection System (RPS). In accordance with the design of the RPS logic, two out of three low AST pressure signals are considered as indicative of a turbine trip. Consequently the Reactor Trip Breakers opened to cause an immediate reactor plant trip. Because of the test in progress, the primary turbine trip system (Auto Stop Oil) was isolated and incapable of providing turbine trip assurance. Necessarily, reliance was placed on the backup emergency turbine trip system involving solenoid valve ET-20.

By design, opening of the Reactor Trip Breakers caused the ET-20 solenoid valve to be electrically energized. The reactor trip also initiated a 30-second delay for opening the output breakers from the Main Generator. Despite the fact that it was energized, the ET-20 solenoid valve failed to open to ensure relief of ETF pressure to keep the turbine steam admission valves closed.

When the AST oil pressure returned to normal, after the momentary perturbation, the AST Interface Valve closed (by design). Since the ET-20 solenoid valve, though energized, did not function, ETF pressure was returned to the pilot valves; this affected the operation of the turbine steam admission valves and initiated reopening of those valves. Depending on the actual configuration of the Stop Valves and Governor Valves, steam may have also been admitted to the turbine through the bypass valve associated with each Stop Valve.

Although various possible individual steam admission valve positions may have existed (including the possible positions of the bypass valves associated with each Stop Valve), the actual configuration apparently was sufficient to admit steam to the turbine at about the same time that the output breakers from the Main Generator opened. The disconnection of the main generator from the grid effectively removed all load resistance from the turbine generator. Consequently, as high energy steam was readmitted to the turbine, the machine experienced an overspeed condition.

At the normal overspeed control setpoint (103% of the normal rated turbine speed of 1800 rpm), the OPC-20-1 and OPC-20-2 solenoid valves were electrically energized. However, neither valve opened to relieve the ETF pressure that was holding the Governor and Intercept Valves open. Consequently the turbine generator unit continued to speed up.

The overspeed condition, in which the turbine reached approximately 2900 rpm, caused several blades in the low-pressure turbine section to separate from the rotor disk, penetrate the 1.25-in.-thick steel turbine casing, and become projectiles from the turbine. Since the Salem turbine generators are located outdoors on the turbine building roof, the projectiles landed on the roof and the ground around the turbine building. No nuclear safety systems were affected by the turbine projectiles.

The resulting eccentric motion of the rotor shaft apparently caused severe vibration in the Main Generator. Consequently the generator hydrogen seals failed and the seal oil lines ruptured. Hydrogen gas (used for generator cooling) and seal oil (used to pressurize the generator hydrogen seals) were released and ignited. A fire erupted in the immediate area of the generator.

When the operators performing the turbine test recognized the situation (about 70 seconds after the reactor trip), they restored the AST system to normal. An operator also manually tripped the turbine to assure that the AST system functioned to open the Interface Valve and relieve the ETF pressure that was keeping the steam admission valves open. The operator actions resulted in finally isolating the turbine from further steam admission. The event duration was about 74 seconds.

In accordance with their emergency plan, the licensee declared the situation as an Unusual Event. The event was later briefly upgraded to an Alert until the licensee determined that turbine projectiles had not affected safety-related systems. All reactor plant systems operated normally, and the reactor was brought to a safe shutdown condition. The fire was extinguished within 20 minutes by a combination of automatically actuated fire suppression systems and rapid response from the on-site fire brigade. No significant personnel injuries occurred. The Unusual Event was terminated in about 3 hours.

The NRC determined that an AIT should be formed to review and evaluate the circumstances and significance of this event. The AIT determined that the proximate cause of the event was the failure of all the backup emergency and overspeed protection trip devices to function because of mechanical binding of the three solenoid valves (Parker-Hannifin Part No. MRFN16MX0834, Westinghouse Style No. 822A848001). The mechanical binding was a result of foreign debris and sludge in the two OPC solenoid valves and foreign debris, rust, and corrosion in the ET-20 solenoid valve.

Several contributing causes and precursor events were identified. The principal findings included the determination that there was no preventive maintenance performed on these valves since installation, and the periodic operational testing of the valves was insufficient to effectively verify the hydraulic performance of each device. Further, by design, the majority of the automatic turbine trip features were bypassed when the mechanical trip testing procedure was performed. In this configuration the turbine trip capability is principally dependent on the proper functioning of a single backup emergency turbine trip solenoid valve, ET-20.

The AIT determined that the event was preventable. In an earlier Licensee Event Report (LER), the licensee committed to replace the ET-20, OPC-20-1, and OPC-20-2 solenoid valves in Unit 2 after discovering on Sept. 10, 1990, that similar components in Unit 1 were defective. An opportunity was available in May 1991 to make these replacements. However, the work was deferred to the planned January 1992 refueling outage by a management decision that may have been caused by a deficiency in commitment tracking. Additionally, on Oct. 20, 1991, operators and their supervisors permitted turbine startup without resolving a turbine system test discrepancy which indicated that the turbine overspeed protection system was not functioning properly.

The AIT reported that the licensee's actions subsequent to the event were effective and correct. The reactor and safety-related systems operated normally and functioned as designed. No radiological release occurred. No safety injection was required. The operators were well trained and qualified and effectively followed the Emergency Operating Procedures for a reactor trip. The reactor was safely stabilized and brought to hot shutdown and then cold shutdown without incident. The licensee correctly classified the event in accordance with the Emergency Classification Guide and made all the required notifications and reports.

Senior management representatives responded immediately to the site and initiated actions to organize, control, and direct event investigation and recovery efforts effectively while keeping the NRC informed. These actions included protecting the scene and configuration for review by the NRC and the licensee's Significant Event Response Team (SERT). The SERT was well trained and qualified and effectively analyzed the occurrence in accordance with well established and recognized event investigation techniques.

The site fire brigade was well trained and equipped. The fire brigade was effective in controlling the fire and mitigating further damage to the facility. All automatic fire suppression systems operated as designed. The flammable materials (hydrogen gas and seal oil) were effectively controlled and isolated to eliminate fuel flow to the fire.

The AIT Assessment made the following points:

> The licensee missed valuable opportunities to prevent the Unit 2 turbine generator failure. A Salem Unit 1 LER dated September 20, 1990 identified failed turbine trip solenoid valves. Insufficient priority and importance was assigned to the verification of operability and replacement of the solenoid valves at Salem Unit 2. Due to the failure to recognize and track the completion of the LER commitment, the licensee elected to defer replacement until the planned refueling outage in January 1992 in lieu of an earlier opportunity during a planned outage in May, 1991.
>
> During the Unit 2 turbine generator startup on October 20, 1991, operators identified an apparent problem with the OPC system, which may have been an indicator of OPC solenoid valve failures. However, several operations personnel, including licensed operators, a shift supervisor, a senior shift supervisor, and a senior operations engineer failed to react appropriately to the problem by assuring proper resolution in accordance with the normal conduct of operations.
>
> Events in 1985 and 1990 at Ginna, in 1988 at Crystal River Unit 3, and the 1990 Salem Unit 1 event were all examples of occurrences involving failed turbine trip solenoid valves that may have been poorly communicated or insufficiently regarded. Further, the NRC issued Generic Letter 91-15, *Operating Experience Feedback Report, Solenoid-Operated Valve Problems in U.S. Reactors,* and the associated NUREG-1275, Volume 6, on September 18, 1991. This report identified several solenoid valve problems, including applications in turbine trip control systems. The Generic Letter did not require any specific response or action, but it did advise licensees to review the information and consider actions to avoid similar problems. The AIT found no indication that the licensee had directed any attention or priority to assessing the implications of this information (relative to turbine control systems) as of the date of this occurrence.

Operating U.S. Power Reactors

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

This update, which appears regularly in each issue of *Nuclear Safety,* surveys the operations of those power reactors in the United States which have been issued operating licenses. Table 1 shows the number of such reactors and their net capacities as of Sept. 30, 1991, the end of the three-month period covered in this report. Table 2 lists the unit capacity and forced outage rate for each licensed reactor for each of the three months covered in each report and the cumulative values of these parameters at the end of the covered quarter since the beginning of commercial operation. The information for this table was obtained from the Nuclear Regulatory Commission (NRC) Office of Information Resources Management. The Maximum Dependable Capacity (MDC) Unit Capacity (in percent) is defined as follows: (Net electrical energy generated during the reporting period × 100) divided by the product of the number of hours in the reporting period and the MDC of the reactor in question. The forced outage rate (in percent) is defined as: (The total number of hours in the reporting period during which the unit was inoperable as the result of a forced outage × 100) divided by the sum (forced outage hours + operating hours).

Table 3 and Fig. 1 summarize the operating performance of the U.S. power reactors during the three months covered by this report (July, August, and September 1991) and for the years 1989 and 1990.

In addition to the tabular data, this article discusses other significant occurrences and developments that affected licensed U.S. power reactors during this reporting period. It includes, but is not limited to, changes in operating status, regulatory actions and decisions, and legal actions involving the status of power reactors. We do not have room here for routine problems of operation and maintenance, but such information is available at the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20555.

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

Table 1 Licensed U.S. Power Reactors as of Sept. 30, 1991

Status	No.	Capacity,[a] MW(e) (net)
In commercial operation[b]	112	100 234
In power ascension phase[c]	0	0
Licensed to operate at full power	112	100 234
Licensed for fuel loading and low-power testing[d]	0	0

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

[b]Excludes Dresden 1 (DER = 200), Humboldt Bay (DER = 65), Three Mile Island 2 (DER = 906), LaCrosse (DER = 50), and Fort St. Vrain (DER = 330), all of which have operating licenses but are shut down indefinitely or permanently.

[c]None at this time.

[d]None at this time.

[a]Oak Ridge National Laboratory.

Table 2 Summary of Operating U.S. Power Reactors as of Sept. 30. 1991[a]

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		July	Aug.	Sept.	Cumulative (lifetime) thru 9/30/91	July	Aug.	Sept.	Cumulative (lifetime) thru 9/30/91
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	95.0 102.4	99.2 102.7	98.0 100.7	59.0 70.3	2.6 0.0	0.0 0.0	0.0 0.0	12.7 11.9
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	7.4 96.1	95.9 97.2	73.2 97.8	56.2 75.5	45.5 0.0	0.0 0.0	0.0 0.0	16.2 4.1
BIG ROCK POINT, Charlevoix County, Mich. (Consumers Power Co.)	50-155	BWR (GE)	240	72	3/63	95.2	102.5	99.4	61.3	6.7	0.0	0.0	12.0
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	72.8 90.6	82.0 64.5	92.2 23.4	61.7 71.4	1.6 0.0	0.0 11.6	0.0 0.0	12.8 4.0
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 51.0 0.0	0.0 75.2 0.0	0.0 85.4 0.0	33.4 32.7 30.8	100.0 0.0 100.0	100.0 0.0 100.0	100.0 7.7 100.0	55.8 54.4 60.2
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	71.3 91.8	100.2 95.5	71.5 34.3	54.0 50.8	24.4 0.0	0.0 0.0	26.1 0.0	15.7 13.3
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	55.1 89.3	35.3 89.4	3.7 91.7	70.3 68.1	0.0 0.0	0.0 0.0	0.0 0.0	2.9 2.8
CALLAWAY 1, Callaway County, Mo. (Union Electric Co.)	50-483	PWR (West)	3411	1171	12/84	100.6	99.2	99.8	81.7	0.0	0.0	0.0	3.1
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	41.8 99.6	99.8 100.7	100.7 99.9	66.4 68.2	42.6 0.0	0.0 0.0	0.0 0.0	9.5 5.6
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	85.5 99.7	95.5 100.0	91.3 60.9	66.6 67.4	5.8 0.0	2.9 0.0	5.3 34.1	12.0 13.3
CLINTON 1, Clinton, Ill. (Illinois Power Co.)	50-461	BWR (GE)	2894	933	11/87	95.7	96.6	95.7	55.3	0.0	0.0	0.0	14.4
COMANCHE PEAK 1, Glen Rose, Tex. (Texas Utilities Electric Co.)	50-445	PWR (West)	3411	1150	8/90	87.5	92.9	92.9	65.5	3.2	0.0	0.0	11.4
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	95.8 89.6	98.6 21.5	79.2 92.3	66.1 61.6	0.0 0.0	0.0 75.1	13.7 0.0	6.8 13.3

(Table continues on the next page.)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Com-mercial opera-tion date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		July	Aug.	Sept.	Cumu-lative (lifetime) thru 9/30/91	July	Aug.	Sept.	Cumu-lative (lifetime) thru 9/30/91
COOPER, Nemaha County, Nebr. (Nebraska Public Power District)	50-298	BWR (GE)	2831	778	7/74	91.7	84.6	76.0	62.9	0.0	0.0	0.0	4.5
CRYSTAL RIVER 3, Crystal River, Fla. (Florida Power Corp.)	50-302	PWR (B&W)	2560	825	3/77	93.6	81.6	95.5	58.3	0.0	5.1	0.0	19.2
DAVIS-BESSE 1, Ottawa County, Ohio (Toledo Edison Co.)	50-346	PWR (B&W)	2772	906	7/78	97.3	71.3	0.0	47.4	0.0	0.0	0.0	25.4
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	100.3 99.1	99.4 92.6	96.6 0.0	74.8 77.1	0.0 0.0	0.0 0.0	0.0 0.0	3.9 5.1
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	51.3 45.7	63.4 38.2	83.5 8.9	58.5 57.4	15.1 0.0	19.7 16.1	1.5 0.0	11.0 11.3
DUANE ARNOLD, Cedar Rapids, Iowa (Iowa Electric Light & Power Co.)	50-331	BWR (GE)	1593	538	2/75	90.0	94.0	95.3	57.8	0.0	0.0	0.0	13.3
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	95.9 98.7	85.7 93.7	98.4 99.0	73.1 82.3	0.9 0.0	9.7 3.7	0.0 0.0	7.2 4.4
FERMI-2, Newport, Mich. (Detroit Edison Co.)	50-341	BWR (GE)	3292	1093	1/88	97.7	99.4	87.7	59.1	0.0	0.0	0.0	9.9
FITZPATRICK, Oswego, N. Y. (Power Authority of State of N. Y.)	50-333	BWR (GE)	2436	821	7/75	0.0	34.0	101.0	65.6	100.0	59.7	0.0	12.3
FORT CALHOUN, Washington County, Nebr. (Omaha Public Power District)	50-285	PWR (CE)	1420	478	6/74	97.3	87.7	38.6	68.3	0.0	1.8	60.4	3.8
GINNA, Ontario, N. Y. (Rochester Gas & Electric Corp.)	50-244	PWR (West)	1520	490	7/70	99.8	91.7	98.2	74.2	0.0	3.0	0.0	6.1
GRAND GULF 1, Port Gibson, Miss. (Mississippi Power & Light Co.)	50-416	BWR (GE)	3833	1250	7/85	88.4	71.0	101.3	74.2	12.7	23.6	0.0	6.6
HADDAM NECK, Haddam Neck, Conn. (Connecticut Yankee Atomic Power Co.)	50-213	PWR (West)	1825	582	8/67	97.5	95.9	98.7	75.5	0.0	0.0	0.0	5.9
HATCH 1 and 2, Baxley, Ga. (Georgia Power Co.)	50-321 50-366	BWR (GE) BWR (GE)	2436 2436	777 795	12/75 9/79	98.5 96.4	88.7 98.2	39.8 99.0	63.5 64.4	0.0 0.0	8.2 0.0	21.2 0.0	13.1 7.6

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, % July	MDC unit capacity, % Aug.	MDC unit capacity, % Sept.	MDC unit capacity, % Cumulative (lifetime) thru 9/30/91	Forced outage rate, % July	Forced outage rate, % Aug.	Forced outage rate, % Sept.	Forced outage rate, % Cumulative (lifetime) thru 9/30/91
HOPE CREEK, Salem, N. J. (Public Service Electric & Gas Co.)	50-354	BWR (GE)	3293	1067	12/86	100.2	99.0	100.7	79.6	0.0	0.0	0.0	5.8
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	31.5 101.5	72.3 97.4	100.2 101.6	60.1 55.2	6.4 0.0	13.7 1.1	0.0 0.0	7.4 15.4
KEWAUNEE, Carlton, Wis. (Wisconsin Public Service Corp.)	50-305	PWR (West)	1650	535	6/74	100.9	99.9	100.6	81.8	0.0	0.0	0.0	2.4
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	100.3 101.9	99.3 98.4	103.0 48.5	58.2 63.4	0.0 0.0	0.0 0.0	0.0 41.1	7.8 13.5
LIMERICK 1 and 2, Pottstown, Pa. (Philadelphia Electric Co.)	50-352 50-353	BWR (GE) BWR (GE)	3293 3293	1055 1055	2/86 1/90	92.5 98.1	96.0 98.2	98.3 99.8	68.4 75.4	0.0 0.0	0.0 0.0	0.0 0.0	4.0 6.9
MAINE YANKEE, Lincoln County, Maine (Maine Yankee Atomic Power Co.)	50-309	PWR (CE)	2560	790	12/72	103.9	103.9	92.8	71.8	0.0	0.0	0.0	7.6
McGUIRE 1 and 2, Cowans Ford Dam, N. C. (Duke Power Co.)	50-369 50-370	PWR (West) PWR (West)	3411 3411	1180 1180	12/81 3/84	97.5 86.0	97.4 98.5	61.7 93.2	61.2 72.2	0.0 6.3	0.0 0.0	0.0 3.6	12.5 8.1
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	0.0 66.0 78.1	3.2 25.6 0.0	86.7 59.8 0.0	70.8 65.9 70.7	0.0 20.8 20.3	81.6 69.6 100.0	7.4 36.4 100.0	10.2 14.7 13.4
MONTICELLO, Monticello, Minn. (Northern States Power Co.)	50-263	BWR (GE)	1670	545	6/71	98.1	92.5	100.3	72.0	0.0	5.8	0.0	3.9
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	60.6 92.4	90.8 38.7	78.8 5.8	54.8 46.6	0.0 0.0	0.0 60.5	14.2 86.5	25.3 25.5
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	52.3 98.7	93.6 98.5	99.5 70.1	64.8 75.8	41.6 0.0	3.4 0.0	0.0 23.4	12.5 6.3
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	95.0 96.4 94.7	0.0 100.0 99.8	0.0 99.7 98.9	69.3 70.6 71.4	0.0 0.0 2.0	0.0 0.0 0.0	93.5 0.0 0.0	11.2 9.7 10.6

(Table continues on the next page.)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		July	Aug.	Sept.	Cumulative (lifetime) thru 9/30/91	July	Aug.	Sept.	Cumulative (lifetime) thru 9/30/91
OYSTER CREEK, Oyster Creek, N. J. (Jersey Central Power & Light Co.)	50-219	BWR (GE)	1930	650	12/69	55.3	73.0	91.9	54.6	30.8	11.0	0.0	11.6
PALISADES, Covert Township, Mich. (Consumers Power Co.)	50-255	PWR (CE)	2200	805	12/71	62.0	103.2	106.5	49.3	20.0	0.0	0.0	31.9
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	101.5 101.4 100.6	101.2 61.0 99.0	62.5 101.2 89.2	51.4 67.6 65.9	0.0 0.0 0.0	0.0 33.1 3.1	0.0 0.0 9.5	21.3 7.5 8.4
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	90.8 71.5	88.4 94.1	97.2 37.5	50.7 52.9	0.6 21.0	6.8 0.0	0.0 0.0	14.5 12.7
PERRY 1, Perry, Ohio (Cleveland Electric Illuminating Co.)	50-440	BWR (GE)	3579	1205	11/87	89.7	97.9	100.3	69.0	2.5	0.0	0.0	7.5
PILGRIM 1, Plymouth, Mass. (Boston Edison Co.)	50-293	BWR (GE)	1998	655	12/72	0.0	28.4	96.4	47.1	0.0	0.0	0.0	12.4
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 12/72	100.8 100.6	99.6 99.3	99.6 88.6	74.7 81.9	0.0 0.0	0.0 0.0	0.0 0.0	1.7 1.1
PRAIRIE ISLAND 1 and 2, Red Wing, Minn. (Northern States Power Co.)	50-282 50-306	PWR (West) PWR (West)	1650 1650	530 530	12/73 12/74	97.2 101.1	86.7 101.3	103.1 101.2	81.2 85.3	0.0 0.0	13.4 0.0	0.0 0.0	5.5 3.0
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	92.4 81.7	94.5 94.4	100.7 53.7	64.9 64.6	0.0 10.2	0.0 0.0	0.0 39.3	5.7 8.1
RANCHO SECO, Sacramento County, Calif. (Sacramento Municipal Utility District)	50-312	PWR (B&W)	2772	918	4/75	0.0	0.0	0.0	33.7 7.0	0.0	0.0	0.0	42.7
RIVER BEND 1, St. Francisville, La. (Gulf States Utilities Co.)	50-458	BWR (GE)	2894	934	6/86	97.3	97.8	67.9	69.6 0.0	0.0	0.0	0.0 3.0	7.3
ROBINSON 2, Hartsville, S. C. (Carolina Power & Light Co.)	50-261	PWR (West)	2200	700	3/71	104.9	91.0	73.7	61.8	0.0	9.1	3.2	15.1
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	97.7 98.2	85.1 96.9	60.4 94.3	56.7 57.9	0.0 0.0	0.0 0.0	16.5 0.0	21.8 22.2

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, % July	MDC unit capacity, % Aug.	MDC unit capacity, % Sept.	MDC unit capacity, % Cumulative (lifetime) thru 9/30/91	Forced outage rate, % July	Forced outage rate, % Aug.	Forced outage rate, % Sept.	Forced outage rate, % Cumulative (lifetime) thru 9/30/91
SAN ONOFRE 1, 2, and 3, Camp Pendleton, Calif. (Southern California Edison Co.)	50-206 50-361 50-362	PWR (West) PWR (CE) PWR (CE)	1347 3410 3410	436 1070 1080	1/68 8/83 1/84	82.3 100.8 101.6	83.1 50.7 99.5	84.7 0.0 100.5	51.0 69.3 70.4	0.0 0.0 0.0	0.0 0.0 0.0	0.0 0.0 0.0	19.1 7.1 7.4
SEABROOK 1, Seabrook, N. H. (Public Service Co. of New Hampshire)	50-443	PWR (West)	3411	1150	8/90	64.1	0.0	0.0	70.8	17.6	0.0	0.0	10.5
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	97.4 97.0	97.8 96.5	87.0 96.9	47.9 51.0	0.0 0.0	0.0 0.0	0.0 0.0	41.7 36.8
SHEARON HARRIS 1, Bonsal, N. C. (Carolina Power & Light Co.)	50-400	PWR (West)	2775	900	5/87	100.0	99.4	100.7	74.2	0.0	0.0	0.0	4.1
SOUTH TEXAS 1 and 2, Bay City, Tex. (Houston Lighting and Power Co.)	50-498 50-499	PWR (West) PWR (West)	3800 3800	1250 1250	8/88 6/89	98.3 98.1	96.1 97.9	87.6 41.2	58.8 66.3	0.0 0.0	0.0 0.0	9.4 0.0	14.4 15.1
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	88.7 102.4	99.0 101.8	94.5 98.7	75.7 84.7	5.7 0.0	0.0 0.0	3.7 0.0	4.1 5.3
SUMMER 1, Broad River, S. C. (South Carolina Electric & Gas Co.)	50-395	PWR (West)	2775	900	1/84	99.4	98.6	63.7	71.2	0.0	0.0	0.0	6.8
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	91.2 79.6	91.2 62.1	95.9 54.7	58.6 58.0	0.0 9.9	0.0 14.2	0.0 28.9	19.6 15.4
SUSQUEHANNA 1 and 2, Berwick, Pa. (Pennsylvania Power & Light Co.)	50-387 50-388	BWR (GE) BWR (GE)	3293 3293	1065 1065	6/83 2/85	96.9 100.3	75.9 62.2	97.9 100.5	72.1 76.9	1.8 0.0	0.0 30.6	0.0 0.0	7.9 6.1
THREE MILE ISLAND 1, Three Mile Island, Pa. (GPU Nuclear Corporation)	50-289	PWR (B&W)	2772	906	12/78	83.2	95.4	85.4	48.7	7.0	0.0	0.0	44.9
TROJAN, Columbia, Oreg. (Portland General Electric Co.)	50-344	PWR (West)	3411	1130	5/76	0.0	0.0	0.0	54.5	0.0	0.0	0.0	13.1
TURKEY POINT 3 and 4, Dade County, Fla. (Florida Power & Light Co.)	50-250 50-251	PWR (West) PWR (West)	2200 2200	693 693	12/72 9/73	0.0 0.0	0.0 0.0	0.0 0.0	60.2 59.8	0.0 0.0	0.0 0.0	0.0 0.0	12.5 12.0

(Table continues on the next page.)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		July	Aug.	Sept.	Cumulative (lifetime) thru 9/30/91	July	Aug.	Sept.	Cumulative (lifetime) thru 9/30/91
VERMONT YANKEE, Vernon, Vt. (Vermont Yankee Nuclear Power Corp.)	50-271	BWR (GE)	1593	514	11/72	96.0	96.9	75.6	73.3	0.0	0.0	0.0	5.6
VOGTLE 1 and 2, Waynesboro, Ga. (Georgia Power Co.)	50-424 50-425	PWR (West) PWR (West)	3411 3411	1157 1157	6/87 5/89	99.0 99.6	96.9 97.9	37.7 95.7	81.2 82.7	0.0 0.0	0.0 0.0	0.0 0.0	7.3 2.5
WASHINGTON NP 2, Richland, Wash. (Washington Public Power Supply System)	50-397	BWR (GE)	3323	1100	12/84	0.0	0.0	0.0	56.5	100.0	100.0	99.8	13.2
WATERFORD 3, Taft, La. (Louisiana Power & Light Co.)	50-382	PWR (CE)	3410	1104	9/85	97.0	94.7	100.6	78.0	0.0	4.6	0.0	4.4
WOLF CREEK 1, Burlington, Kans. (Kansas City Power & Light Co.)	50-482	PWR (West)	3411	1170	9/85	98.0	94.9	49.1	77.2	0.0	0.0	0.0	3.5
YANKEE ROWE, Rowe, Mass. (Yankee Atomic Electric Co.)	50-29	PWR (West)	600	175	11/60	92.3	81.7	97.4	74.7	0.0	6.3	0.0	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	89.9 80.8	93.0 84.6	95.6 78.7	57.0 60.9	0.0 6.0	0.0 0.0	0.0 0.0	16.6 15.0

[a]The information in this table is obtained from NRC Publication NUREG-0020, Vol. 15, Nos. 5, 6, and 7.

Table 3 Power Generation During the Third Quarter of 1991

Power generation	1989	1990	July	August	September	Year-to-date
Gross electrical, MW(e)h	555 666 518	605 169 082	63 756 511	61 307 120	54 419 134	488 156 898
Net electrical, MW(e)h	528 204 992	575 991 274	60 783 346	58 473 566	51 841 211	465 126 227
Average unit factors, %						
Service	68.2	71.1	86.3	84.3	77.2	74.9
Availability	68.5	71.1	86.3	84.3	77.2	74.9
Capacity						
MDC	63.3	67.0	81.1	79.1	73.6	71.1
DER	61.9	65.5	79.3	77.3	72.0	69.5
Forced outage rate	11.2	9.7	7.1	9.0	9.0	10.3

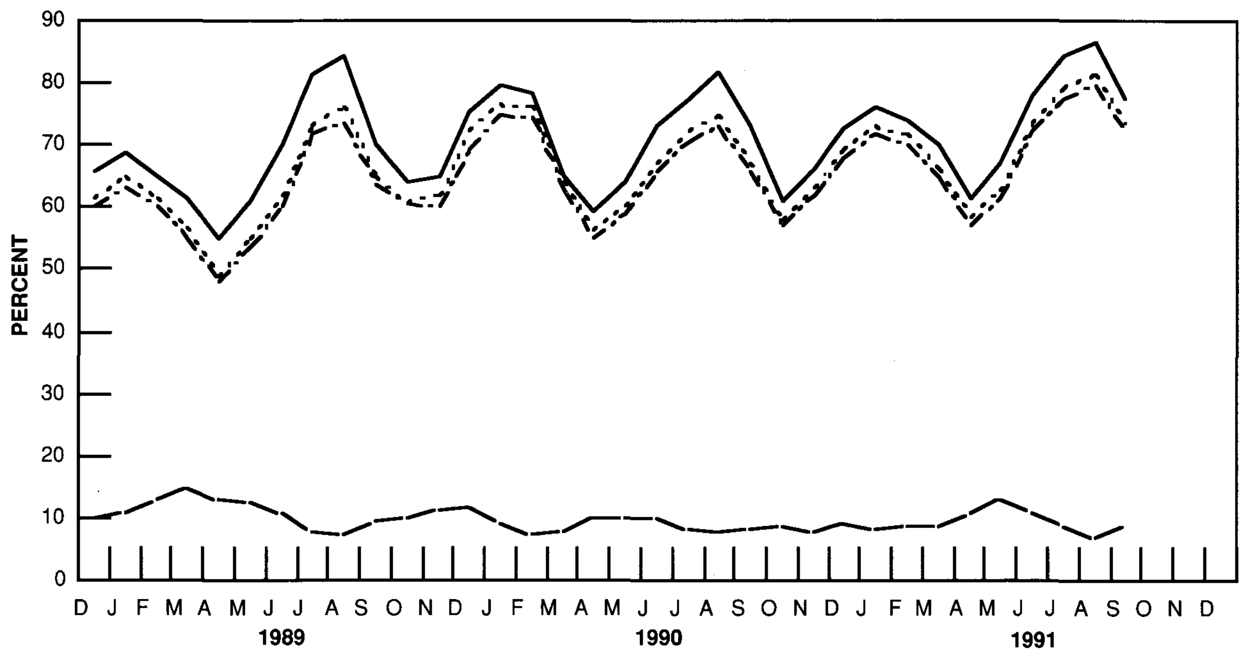

Fig. 1 Average unit availability, capacity factors, and forced outage rate. ——, availability factor. ...., MDC capacity factor. .—.—, DER capacity factor. ———, forced outage rate. Data through February 1990 are obtained from NUREG-0200; data for the remainder of 1990 were obtained from the NRC Office of Information Resources Management. 1991 data are obtained from the magnetic-media version of NUREG-0200.

The reader's attention is also called to the regular features "General Administrative Activities," which deals with more general aspects of regulatory and legal matters, "Waste and Spent Fuel Management," which covers legislative, administrative, and technical matters related to the back end of the fuel cycle and to management of radioactive wastes in general.

REACTOR VESSEL EMBRITTLEMENT AT YANKEE ROWE LEAVES FUTURE IN DOUBT

On the basis of testimony at a public hearing,[1] an affirmation hearing,[2] and a Senate Subcommittee hearing[3] during July and early August 1992, the question at

Yankee Rowe was not whether reactor vessel embrittlement had occurred, but whether the plant could continue to operate safely with the level of embrittlement that has occurred. According to NRC Chairman I. Selin, "All parties agreed that the major source of concern was the extent to which the pressure vessel might have become brittle through bombardment by high energy neutrons." The chairman continued by saying that trouble could arise if a specific set of events led to a pressurized thermal shock (PTS), caused by a massive loss of reactor coolant that would cause the emergency core cooling systems to flood the vessel with cold water (causing thermal shock), while the vessel remained at high pressure. The PTS would cool the inside of the vessel while the outside remained at high temperature, which could lead to extensive failure of the vessel by brittle fracture if the neutron-induced embrittlement were severe enough. The customary measure of the degree of embrittlement is the temperature at which the ductility of the material goes to zero, called the null ductility transition. Initially this temperature is very low, well below any temperature the vessel could reach in operation, but as the total fast-neutron fluence increases with time (i.e., the total number of fast neutrons per unit volume to which the material has been exposed), the null ductility transition temperature can rise to above room temperature so that cold water could locally shift the material to the brittle state. On the basis of that information, Selin said the Commission had agreed that "if there were particular pressure and temperature conditions and if there were flaws of particular size and orientation in the most vulnerable part of the vessel, it might rupture under this pressurized thermal shock."[4]

Selin, when asked to tell, "in plain English," the likelihood of an accident occurring at Yankee Rowe, said that, in order to determine that, two factors must first be analyzed: (1) the likelihood (frequency) of an event actually occurring and (2) the ability (probability) of the pressure vessel to withstand such a shock. "If an event happens, the likelihood is high that the vessel will crack," said Selin, which indicates that, so far, NRC had been banking on the fact that there was only a small probability that a PTS event would occur. Selin said further that he was convinced that the NRC staff acted properly but that if new information arose he would not hesitate to shut down the plant before its scheduled April 1992 outage.[5]

The Union of Concerned Scientists (UCS) and the New England Coalition on Nuclear Pollution (NECNP), whose request for an immediate shutdown of the plant was denied, claimed that the plant is in violation of NRC requirements for reactor pressure vessel integrity and that a safety assessment of the plant performed in August 1990 was not conducted in a sufficiently thorough manner.[6] According to D. Curran, legal counsel for UCS, "it is only the allegedly low probability of an accident occurring that is allowing the plant to stay open." [Exact values for the probability of brittle failure are difficult to come by. The conditional probability of such a failure (i.e., the probability of failure *given the occurrence of a PTS event* ranged from an NRC staff estimate of 0.03 to an estimate of 10-5 by the staff at Yankee Rowe). The NRC had also commissioned a study of this matter at the Oak Ridge National Laboratory (ORNL); a draft report by ORNL reported that this study resulted in a conditional probability of 0.08 to 0.09. NRC staff asserted that they stood by their figures and noted that their calculations were based on very conservative estimates. NRC noted further that "we don't know precisely if there are any flaws in the beltline welds, but the flaw size and distribution of those we have looked at is conservative, compared to other reactors we've looked at."]

The conservatisms in the NRC and ORNL estimates are due to the uncertainties caused by the lack of available data on the status of welds in the reactor pressure vessel. It was precisely this lack of data that convinced the UCS that the plant should be shut down. The licensee stated that they were currently working on developing technology and equipment to test samples from the circumferential weld and that it should be ready to deploy the new equipment by April 1992, when the reactor was scheduled to be shut down for refueling and maintenance.

Yankee was to submit its plan to resolve the uncertainties to the NRC by Aug. 26, 1991 (Ref. 7). In July the NRC Commission saw ". . . no need or benefit in ceasing plant operation before the uncertainties could be addressed," which meant, until the techniques and equipment had been developed to examine welds and search for flaws in the pressure vessel, the Commission saw no need, on the basis of its current assessment, to shut down the plant. The Commission, however, also added that, "In no event will plant operation continue beyond April 15, 1992, without first having resolved these uncertainties."

After hearing NRC staff testimony regarding the validity of its assessment that the plant is safe, Selin broke in: "This is not a game—it's not just a question of sensitivity analyses. Ask yourselves where we could be wrong. Put yourselves in both sets of shoes, and look at areas like axial welds, high reference temperatures, and pressure vessel shocks."

"The issue at stake is aging," said Selin, "and it will be, over and over again."

After the affirmation vote, Selin was asked what impact the Commission's decision on Yankee Rowe would have on the future of other nuclear plants as they come up for relicensing. He responded that, as a result of the deliberations over Yankee Rowe, new criteria had been established setting specific standards for pressurized thermal shock boundaries at 1 in 100 000 reactor years.

The Commissioners found themselves under fire from the House Subcommittee on Energy and the Environment, chaired by Rep. P. H. Kostmayer (D-Pa.), at a hearing called for the express purpose of reviewing NRC's decision permitting Yankee Rowe to continue operating.[8] Kostmayer began by noting that the Commission made its decision that Yankee is safe for the time being, even though it lacked several critical pieces of information necessary to determine the ability of the reactor pressure vessel to resist cracking under specific conditions. Kostmayer also noted the Commission's own conclusion that there was a one-in-ten probability that the reactor vessel would crack if the emergency core cooling system were activated. Given these "alarming" probability estimates, Kostmayer demanded to know whether continued operation of the plant was warranted. He also argued that the NRC was taking an unnecessary gamble with the health and safety of the people; that the Commission seemed to be saying, "We are willing to allow plants to operate based on possibly flawed probability estimates when we fully admit that we could eliminate any uncertainties by requiring shutdown until all necessary tests have been completed." Kostmayer further added that, "Because there are uncertainties at this plant, we are willing to grant the licensee extra latitude because the situation probably isn't as bad as our conservative estimates make it seem."

Rep. J. Olver (R-Mass.), in whose district the plant is located, urged that, since the scientists and engineers can only obtain the information they need when the plant is shut down, it should perhaps be shut down immediately. J. Swift, a member of the Massachusetts State Senate who said that she spoke as a representative of the people of Rowe (she lives within the emergency evacuation zone of the plant), stated that Rowe is entitled to a "fair evaluation" and that any decision should be based on rational, objective, and scientific criteria. But that has not been the case with Rowe, said Swift, who argued that the larger question of the national energy policy has been allowed to obscure the issue. Swift said that she supports NRC's decision to keep Rowe open.

Selin then presented a statement explaining the NRC's position. Thereupon Kostmayer led off the questioning: "Is it correct to assume that when the plant is shut down in April (1992) for refueling, it will probably be shut down for a long time?" Selin answered in the affirmative.

"And the only way to address the uncertainties is to shut the plant down, correct?" Kostmayer continued. Selin again agreed, and Commissioner K. C. Rogers added, "The equipment to perform the tests is not currently ready, and still has to be tested after it is designed."

"Yes, and there are large uncertainties that can't be resolved other than by shutting it down, right?" repeated Kostmayer, and went on to ask: "And there are questions that should be answered as soon as possible, aren't there?" Selin agreed, so Kostmayer homed in: "Then why not shut it down now?"

In response, Selin explained that, although it was true that answers to the uncertainties could not be obtained until the plant was inoperative, "certain other things have to happen" before it would do any good to perform the tests (namely, that equipment and procedures would need to be developed to perform the complicated tests).

Selin further argued that chances are minuscule that the tests would actually reveal the vessel to be as susceptible to breaking as the uncertainties could lead one to guess. "First of all," said Selin, "the flaw has to be just the right kind of flaw. It has to be perpendicular to the inside wall of the plate, it has to be small, and it has to go through both the stainless-steel and carbon-steel walls of the pressure vessel." Somewhat plaintively, Selin added: "The reality can't be any worse than the calculations."

Rep. C. Thomas (R-Wyo.) wanted to know what, specifically, NRC has asked the licensee to do before the scheduled outage. Selin's response was that the licensee would first try to determine whether, if a break did occur, it would be large or small, a question that, in the Commission opinion, can be answered. Second, Selin said, the licensee must be able to keep at least two of the main cooling pumps operating in the event of a loss of coolant accident (LOCA), which would help mitigate the danger.

Reps. G. Darden (D-Ga.) and J. J. Rhodes (R-Ariz.) both said that, although they were not experts on nuclear energy, they were most interested in establishing that the Commission had performed its duty in a complete and responsible manner. Both congressmen congratulated the Commissioners for quickly and comprehensively addressing the issue.

A. C. Kadak, president and CEO of Yankee Atomic Electric Company (the licensee), then argued that Yankee was safe to operate because: (1) Analyses of the vessel have repeatedly demonstrated that the plant is safe to operate, (2) the reactor has inherent safety features that make it less likely to have a PTS event than other plants, and (3) NRC and the Advisory Committee on Reactor Safeguards (ACRS) have concluded that the plant is safe to operate for an additional cycle. He added that, "We would not be operating the plant today if we did not believe it to be safe to do so."

D. Curran, legal counsel for UCS, faulted the Commission for "reversing prior precedents regarding the interpretation of Appendix G to 10 CFR Part 50, which requires testing of the pressure vessel." By claiming that one section of the rule did not apply to Yankee Rowe, Curran said, the Commission had incorrectly justified interim operation of the plant. He also complained that the Commission's denial to grant the petitioners request for an adjudicatory hearing proved that NRC did not provide an open and fair opportunity for public participation. The petitioners, he said, would pursue their request for an adjudicatory hearing with a federal court of appeals.

Later in the quarter, as required by the NRC order, the utility submitted a short-term plan for reducing the possibility of vessel rupture in the event of a PTS event. The plan asserts that it is indeed possible to reduce the probability of vessel failure by a factor of 20 (Ref. 9). The letter from Yankee Atomic said that it had completed its evaluation and that "our analysis shows that if we implement our suggested proposal of continuing to operate two of four main coolant pumps in the event of a small break loss of coolant accident, we can reduce the vessel failure probability by a factor of 20." The letter went on to say that the utility was in the process of preparing procedures and making small design changes to set up the main coolant pump running procedure and that, once the NRC had reviewed the analysis, the licensee was prepared to implement the new measures promptly. By keeping two main pumps operating, the letter said, water circulation within the reactor vessel would mitigate the danger of a PTS event.

The analyses performed by Yankee have shown, the report asserts, that the "enhanced mixing" of colder safety injection water with warmer main coolant from running pumps results in lower thermal stresses in the reactor vessel wall and therefore reduces the chances of a through-wall crack.

Yankee said that their analysis was conservatively developed and that the results demonstrate that the main coolant pumps would operate throughout an event in the containment environment that would arise from a LOCA.

CONTROVERSY SURROUNDING SHOREHAM CONTINUES

The controversy surrounding the $5.5 billion construction of the Shoreham Nuclear Power Station in Long Island, N.Y., continued. In mid-June 1991 the NRC approved a license to dismantle Shoreham.[10] Although this ruling undoubtedly put a damper on the hopes of the Administration and DOE to reopen Shoreham, it did not stop them altogether from trying to do so.[11]

The U.S. Department of Justice (DJ) moved in mid-June 1991 to force a review of the environmental consequences of dismantling the New York plant. At the request of the Bush Administration, DJ filed papers at the Federal Court of Appeals in Washington, D.C., seeking to stay the granting of Shoreham's "possession only" license. The Administration, which has long favored the operation of Shoreham, argued that an environmental study should compare the use of electricity power sources, such as fossil fuels, against the atmospherically cleaner energy that Shoreham could provide. By the JD action the Administration hoped at least to delay the dismantling of Shoreham, during which it could try to persuade the parties involved.

The DOE, siding with the Administration, is on record as believing that the reactor at Shoreham is safe and should be allowed to operate. The DOE had previously unsuccessfully requested the NRC to refuse a "possession only" license. DOE, along with other Shoreham supporters, stated in court that, "It is clear that the approval of the possession-only license [would be] likely to result in actions being taken at the plant which could render moot a meaningful consideration of alternatives to decommissioning or limit the methods of decommissioning available to the NRC." The NRC had previously decided that no environmental impact statement was necessary prior to dismantling the plant.

State officials in New York were unhappy with the JD action; they prophesied increased costs to ratepayers on Long Island as a result of the delay the action would cause. An official of the state agency responsible for dismantling Shoreham said: "Shoreham is never going to operate as a nuclear power plant. All this will do, if it is successful, is further delay the decommissioning and raise the rates for the customers. The JD should have better things to do than to continue to harass Long Island on an issue that should be dead."

The NRC did not join the JD position, although it did enter an administrative stay that expired on July 19 to allow the court to consider the matter. NRC contended that publication of an EIS considering "resumed operation" of Shoreham as an alternative to decommissioning was not required under the National Environmental Policy Act.[12]

In a brief conclusion, NRC's general counsel stated that, "There is no doubt that the Shoreham experience has been an unfortunate one. It simply adds to the misfortune when those favoring nuclear power attempt so vigorously, as Judge Williams put it, to 'raise the costs of exit from the nuclear industry.' This effort seems 'certainly counter-productive' from petitioners' own point of view and also threatens to increase the costs of electricity for society at large."

NRC PROPOSES TO REVISE RULES GOVERNING FUNDING OF DECOMMISSIONING FOR PREMATURELY SHUTDOWN NUCLEAR POWER PLANTS

The NRC proposed to amend its regulations governing the collection of funds to pay for the decommissioning of nuclear power plants that are shut down before the expected end of their operating lives. Under the proposed rules, the NRC staff would evaluate decommissioning funding plans on a case-by-case basis, taking into account the specific financial and safety situations at each plant.[13] They provide that, for those nuclear power plants which were permanently shut down prior to July 27, 1988, the Commission may approve modifications of the decommissioning plan, including provisions for assuring that adequate funding is available. They also require utilities that permanently shut down their nuclear power facilities after July 27, 1988, to make an application to terminate the license accompanied by or preceded by a proposed decommissioning plan. The application must be submitted within 2 years following permanent cessation of operations and, in no case, later than 1 year prior to expiration of the operating license.

The Commission's existing regulations reflect its objective that, at the time of permanent end of operations, sufficient funds must be available to decommission the facility in a manner that protects the public health and safety. The proposed revisions to these requirements reflect: (1) The permanent shutdown (after the July 27,1988, effective date of the final decommissioning regulations) of the Fort St. Vrain (Colorado), Rancho Seco (California), and Shoreham (New York) nuclear power plants; and (2) subsequent Commission guidance to the staff on the appropriate period for collecting funds to compensate for any shortfall of decommissioning funds for these facilities.

As proposed, the following criteria would be used to evaluate funding options for licensees that shut down their nuclear power plants prematurely:

1. All funds needed for decommissioning should be available or guaranteed in external accounts before the start of DECON operations (the removal or decontamination of equipment, structures, and portions of the site contaminated with radioactivity to a level that would permit release of the site for unrestricted use); however, the accumulation of funds during a SAFSTOR period (a period in which the plant would be placed and maintained in a condition that it could be safely stored and decontaminated to levels that would permit release of the site for unrestricted use at a later date) would be permitted.

2. Other factors, such as the number of power plants in a licensee's system that continue to generate revenues, would be considered.

3. Licensees who elect to collect funds in external accounts during the SAFSTOR period would be required to demonstrate financial solvency during the collection process. These licensees would be permitted to continue such collections only until the term of the original operating license expires and would have to have the necessary decommissioning funds available or guaranteed prior to final dismantlement operations.

NRC ORDERS ALL OPERATING LICENSEES TO PARTICIPATE IN NEW EMERGENCY RESPONSE DATA SYSTEM PROGRAM

The NRC also amended its regulations to require all licensed nuclear power facilities (except Big Rock Point and plants that are permanently or indefinitely shut down) to participate in the Emergency Response Data System (ERDS) program. The ERDS will link all nuclear power plants in the United States with the NRC during alerts or emergencies. The action, which became effective Sept. 12, 1991, requires licensees to submit timely and accurate data to NRC on "a limited set of parameters whose values indicate the condition of the plant" during alerts or higher emergencies.[14] NRC stated that the action was intended to "ensure that all licensees establish a definite schedule" for implementation of the ERDS program. [In August 1989 the NRC requested voluntary ERDS participation by the licensees. By January 1991, however, only about half the operating plants voluntarily participated.[15] This new regulation makes participation mandatory.]

The ERDS is a direct electronic data link between computer data systems used by licensees of operating reactors and the NRC operations center during an alert or higher emergency. The ERDS supplements (but does not replace) the emergency notification system (ENS), which is used for voice transmission of information when an alert or higher level emergency is declared at a licensed nuclear power facility.

The Big Rock Point plant was exempted from this regulation because the plant only has five data points available for ERDS—an insufficient number of parameters for effective participation, according to the NRC. In addition, licensees that previously implemented ERDS voluntarily in an acceptable manner were not required to submit an implementation plan.

REACTOR COOLANT PUMP MOTOR INVOLVED IN TRAFFIC ACCIDENT

A spare reactor coolant pump motor from the Surry Power Station, weighing 40 tons and rated at 6000 horsepower, fell onto Interstate 64 near Newport News, Va., in late August 1991 and thus spilled radioactive water onto the highway after the truck carrying the motor failed to clear an overpass.[16] The truck and its load were 15 ft high, whereas the overpass clearance was only 14 ft and 7 in. The box fell partway off the truck onto the pavement and leaked water onto the shoulder of the road. The crash was caused by a wrong route assignment according to a Virginia Power Company transportation department spokesperson.

The spill was quickly contained in a 200-ft puddle. NRC officials said that the radiation levels were far below the level considered harmful to humans and that the spill posed no threat to residents living near the accident scene or to people passing by in cars. No one was injured in the accident, according to a police report.

A crane lifted the motor and its container back onto the truck, and it was then carried back to the Surry plant.

Virginia Power, the plant's owner, also removed asphalt and concrete from where the radioactive water spilled and took it back to Surry to be treated as low-level waste.

Virginia Transportation Commissioner R. D. Pethtel ordered an immediate change in the way routes are designated on thousands of permits issued for oversized vehicles.

FULL TERM OPERATING LICENSE FOR SAN ONOFRE PLANT EVALUATED

San Onofre, Unit 1, received a provisional operating license in March 1967 and began commercial operation in January 1968. The licensee applied for a Full Term Operating License (FTOL) in July 1970, but review of this application was deferred by the NRC staff in 1975 (Ref. 17). Thus San Onofre had been operating for 24 years on a temporary license. The NRC appeared to be moving closer to converting that temporary license to an FTOL. In addition, the ACRS recently stated its conclusion that there is reasonable assurance that the San Onofre 1 can continue to be operated at power levels up to 1347 MW(t) under a full-term operating license without undue risk to the health and safety of the public.[18]

Survival of the facility, however, may depend on the California Public Utility Commission (CPUC). For an FTOL, the NRC was requiring $360 million worth of safety improvements at the reactor, which originally cost $89 million to build, and even more safety alterations are said to be needed. The CPUC scheduled hearings on whether to charge ratepayers $125 million for the additional improvements sought by the NRC. Refusal to allow the rate increase could force closure of the power plant.[19]

When evaluating the licensee's request for an FTOL, the NRC took into account nonradiological and radiological impacts, noting that, since 1973 big changes have taken place at San Onofre. Most important, two additional reactors (Units 2 and 3) were constructed adjacent to Unit 1. Although the new construction affected land use, NRC concluded the effects were "minor and local." The licensee received good marks from NRC in terms of its control of the local environment in the vicinity of the site. San Onofre's drainage system is good, and terrestrial resources along the transmission lines have not been damaged.[20]

The California Coastal Commission is evaluating the results of studies conducted to determine the site's effects on the marine environment. Although those studies primarily focus on the marine impact of Units 2 and 3, NRC noted that Unit 1 also impacts the marine environment and should therefore be included in marine analyses. In general, with regard to federally regulated issues, NRC concluded that the site's effects on ocean life were acceptable; additionally, NRC stated that state oversight of the marine impact studies is appropriate. There is one area of possible concern, however: *Chelonia mydas*, the green sea turtle. The power station and the transmission rights of way may have had some impact on the sea turtle population, although NRC staff believes this to be unlikely. A legally protected species, the green sea turtle population near San Onofre will be studied in consultation with the National Marine Fisheries Service.

Radiologically speaking, the NRC also found that San Onofre has kept within the federal limits for release of radioactive effluents: "Radioactive gaseous and liquid effluents during the first 23 years of operation did not exceed regulatory limits." Furthermore, NRC reported that "potential exposures to personnel in unrestricted areas resulting from plant effluents were determined to be within the design objectives contained in 10 CFR part 50." With that analysis in hand, the NRC concluded that the radiological environmental consequences of plant operations for the duration of the proposed FTOL are acceptable. The NRC did not announce when it intends to publish a decision on converting San Onofre's POL to an FTOL, and it remained to be seen whether an amended license would be a factor in the CPUC's determination of whether or not to shut San Onofre down.

THE ACRS COMMENTS ON THE EVALUATION OF RISKS DURING LOW POWER AND SHUTDOWN OPERATIONS

During its Aug. 8–9, 1991, meeting, the ACRS continued its discussion of the NRC staff program to address the risks posed by nuclear power plants during low power and shutdown operations.[21] The ACRS report states, in part:

> We share the staff's concern that this issue needs to be addressed in a thorough and systematic manner and are favorably impressed with the approach being taken. We are encouraged that the industry is also actively pursuing this issue.
>
> There are three aspects of the staff's shutdown risk study that we believe merit comment:
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> 1. The staff was unable to provide us with the information concerning the design of containment equipment hatches that we had requested during our review of NRC Generic Letter 8817 on loss of decay heat removal. We had asked how many plants have hatches that are pressure-seating and could be easily closed if the containment were in danger of being pressurized, as opposed to plants having pressure-opening hatch designs that require essentially full bolting to accomplish sealing under pressure. This appears to us to be an important question that could be answered by referring to available information. A related issue concerns the ability of the licensees to effect closure of their equipment hatches when AC power is not available. The March 1990 loss-of-power event that occurred at Vogtle, Unit 1, demonstrated the importance of this consideration. The NRC staff has stated that these matters will be addressed as part of the shutdown risk study.
>
> 2. One component of the shutdown risk study is the development of two PRAs designed to quantify risks posed by low power and shutdown operations. The two plants, Surry and Grand Gulf, chosen for these studies are among those previously modeled as part of the NUREG-1150 studies. We pointed out to the staff that neither of these plants is a good surrogate for the U.S. population of operating reactors. Surry is one of the few PWRs that has isolation valves in its reactor coolant system which permits the licensee to minimize operation at "mid-loop" conditions. Grand Gulf represents the BWR/6 product line; as such, it is representative of only a small fraction of the total population of operating BWRS.
>
> The staff acknowledged this point, but argued that the review of these plants in the NUREG-1150 effort aids in evaluation of shutdown risk. The willingness of the owner/operators to participate in this study was also a consideration. The degree to which these plants can be considered representative of their surrogate populations will need to be established if the shutdown PRA studies are to be relied on in making regulatory decisions concerning the resolution of this issue.
>
> 3. Another concern deals with the NRC staff's modeling approach for the PRA studies. The staff has a two-pronged effort under way. For the short term, a coarse "screening analysis" using "conservative" assumptions will be performed on a schedule that supports the staff's commitment to provide recommendations by the end of the year on measures to minimize shutdown risk. For the long term, a more complete PRA study will be conducted. The long-term effort will not be complete at least until some time during 1992-93.
>
> The staff's discussion of the conservatism being used in these screening analyses raised concerns with us as to the usefulness of this work. For example, we were told that modeling of human error would be dealt with by assuming that, in most cases, the operator makes the wrong decision in taking action during sequences that could lead to core damage. Since these studies will presumably play some role in the recommendations that the staff will present later this year concerning amelioration of shutdown risk, we caution that PRAs performed in this manner can lead to badly flawed regulatory decisions.
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> Our views on the use of PRA in the regulatory process are further discussed in our report of July 19, 1991, to Chairman Selin. We recommend that the staff carefully consider the comments presented in that report.

ACRS COMMENTS ON GENERIC ISSUES 130 AND 153

At its August 8–9 meeting, the ACRS considered the NRC's proposed resolutions to Generic Issue (GI) 130 (Essential Service Water System Failures at Multi-Unit Sites) and to the NRC staff action plan for GI 153 (Loss of Essential Service Water in LWRs). Its conclusions were sent to the Chairman of the NRC in a letter report that reads, in part:[22]

> Since licensees will be examining their essential service water systems (ESWS) in detail as an important part of their IPE [Individual Plant Examination] used by the staff in its proposed resolution of GI-130 available to licensees.

This information should assist them in carrying out their IPEs. We do not, however, understand your statement that "... using the IPE as our vehicle to resolve this generic issue is not a practical option." It seems to us that, if these licensees do a conscientious job of performing their IPEs and identify and correct vulnerabilities involving their ESWS, resolution of the GI-130 issue can be accomplished on a plant-specific basis within a reasonable time.

We believe that the analysis of GI-130 was extremely conservative with respect to the methodology used to establish 1) the frequency of loss of ESWS and 2) the accident mitigation attributes of the "representative plant" for these plants. This was recognized by your contractor, Brookhaven National Laboratory, on page vi of the Executive Summary of NUREG/CR-5526, where the statement is made that "... the service water-related CDF ... is considered to be essentially upper bound."

The ACRS has historically recommended that PRAs be performed on a best-estimate basis and that conservatism then be added when needed to deal with uncertainty for regulatory purposes. (We most recently discussed this issue in our report of July 19, 1991, to Chairman Selin on the subject of "The Consistent Use of Probabilistic Risk Assessment.") It is clear to us that this principle was not applied to the staff's proposed resolution of GI-130 and is not generally applied by the staff to the cost benefit analysis used for generic issue resolution.

Further, we note that RES has recently developed a Task Action Plan (TAP) for Generic Issue 153, "Loss of Essential Service Water in LWRs." This work represents an expansion of GI-130 to the remaining 99 operating LWRS. The TAP states that the IPEs for the population of operating plants "... may provide information related to the ESW system" and "... may also result in an ESW risk model for each plant, which may be useful for this task. "We fail to see how a meaningful IPE can be performed without a detailed evaluation of a plant's ESWS and the accident sequences that could result from partial or complete loss of ESWS.

We believe that GI-153 is well enough defined that it could be resolved on a plant-specific basis as part of the IPE process, and we recommend that this approach be followed. We believe also that there may be other generic issues at a similar stage of development and suggest that work on their resolution could be deferred until enough IPEs have been received and evaluated to determine if the expenditure of staff resources to deal with them as generic issues is warranted. We would like to be kept informed on this matter.

AUGMENTED INSPECTION TEAM SENT TO WOLF CREEK

The NRC staff in Arlington, Tex., sent a five-member Augmented Inspection Team (AIT) to the Wolf Creek nuclear power plant near Burlington, Kans., to look into the circumstances surrounding a partial loss of offsite power on Sept. 23, 1991, and the subsequent tripping of two reactor coolant pumps, the loss of a circulating water pump, and the loss of service air to the spent-fuel pool transfer canal boot seal.[23] The latter loss allowed water to leak from the spent-fuel pool.

The plant was shut down for refueling when the event occurred; the water leaking from the spent-fuel pool was not discovered until almost 2 hours after the leak began. As a result of the loss of the seal, 6 to 8 ft of water leaked from the top of the spent-fuel pool into the transfer canal and thus rendered the spent-fuel cooling system inoperable.

There was no release of radioactivity to the environment and no danger to the public or plant workers during the event and the spent fuel remained covered with water.

A report of the AIT's findings was expected to be completed by late October 1991.

IAEA EXAMINES OLDER VVER REACTORS

A team of safety inspectors, working under the sponsorship of the International Atomic Energy Agency (IAEA), examined ten older, Soviet-designed nuclear reactors in Czechoslovakia, Bulgaria, and the Soviet Union and came up with a list of more than 1000 recommendations on how to make them safer. The reactors in question are 440-MW VVER pressurized-water reactors (PWRs) without the containment structures to constrain radioactive releases in the event of a major accident, which are prominent features of all Western PWRs. The plants also lack numerous other safety features that are standard in the West. The inspectors outlined their recommendations at the IAEA's 35th annual meeting held in September 1991. The recommendations ranged from a higher level of safety consciousness and more training for operators to tighter management accountability and cleanups for neglected installations. They also called for several specific hardware fixes involving the safety of welds along the reactor's pressure vessels and the reliability of monitoring and control instruments. Because the year-long inspection program was just being completed and analysis and fund-raising were about to begin, IAEA officials were reluctant to make precise estimates about the expected costs to bring the ten reactors to an acceptable level of safety.[24]

EIGHT NEW FINES DURING REPORTING PERIOD

Eight new penalty fines have been levied by the NRC on reactor licensees during the three-month period covered by this report (the third quarter of the year 1991). In each case the affected utility was required to report to the NRC on the causes and proposed corrections of the problem that led to the fine and had 30 days from the date of notification to either pay the penalty or protest its imposition in whole or in part. Each of the eight cases is briefly described here.

Farley 1: Inoperable Auxiliary Feedwater Pump

The NRC staff proposed a \$25 000 civil penalty against Alabama Power Company for alleged violation of NRC requirements at the Farley nuclear power plant located near Dothan, Ala.[25] NRC officials said the problem, which was identified by the plant staff, occurred from May 17 until May 22, 1991, when Unit 1 went from hot standby to power operations over that period of time with its turbine-driven auxiliary feedwater pump (AFWP) flow path blocked as the result of a valve misalignment. The AFWPs are used to provide a backup source of water to the steam generators to provide a heat sink for decay heat removal from the primary coolant loop in the event the main feedwater pumps are unavailable following a shutdown. So that the possibility of common-mode failures will be minimized, PWRs generally contain some combination of electrically driven and steam-turbine-driven AFWPs. At Farley, the ability to deliver auxiliary feedwater during a shutdown would have been degraded with the steam-driven pump out of service.

The NRC said the problem was caused by ineffective procedural controls and communications. The valve had been opened for a test while the reactor was in hot standby and was not properly closed and locked as it should have been at the completion of the test. According to the NRC, the base civil penalty for this type of violation is \$50 000, but it was reduced to \$25 000 in this case because of the company's previous high NRC ratings in periodic evaluations of its operations and the good prior enforcement history of the plant.

Fort St. Vrain: Unplanned Radiation Exposures

A Notice of Violation and Proposed Imposition of Civil Penalty in the amount of \$62 500 was issued on August 15 to Public Service Company of Colorado for violations of NRC radiation-protection requirements at the Fort St. Vrain facility.[26] The action was the result of an incident on June 11, 1991, when a mechanic and a health physics technician received unplanned radiation exposures to their hands. The exposures, which did not approach NRC limits, occurred when these employees unknowingly handled highly radioactive material in the course of decontaminating a shipping cask at the plant.

The NRC action was based on violations of radiation safety requirements, including failure to (1) make proper radiation surveys, (2) issue a radiation work permit, (3) instruct workers on how to minimize exposure, and (4) supply appropriate personnel monitoring equipment. Although the amount of the civil penalty was mitigated by 25% for corrective actions, it was escalated 50% because of the poor past performance of the licensee in this area.

Millstone 1: Blocked Intake Water Filters

The staff of the NRC has cited Northeast Nuclear Energy Company for failing to follow written procedures at Millstone 1, located near Waterford, Conn. The staff assessed a \$50 000 fine for the violation.[27] The procedures require the Shift Supervisor (SS) to shut down all four of the circulating water pumps that send Long Island Sound water into the plant's main condenser whenever the control room indicators show too great a pressure drop across the screens that filter the water as it enters the plant.

On Oct. 4, 1990, a severe storm caused a buildup of seaweed on these screens. Although control room indicators showed an increased pressure drop across the screens, the SS tripped only two of the pumps because he believed the indicators to be incorrect. This caused a partial collapse of three of the screens, which allowed seaweed into the intake bays and thus clogged a service water strainer. There also was a potential for clogging the emergency service water strainers.

In a letter to Northeast Nuclear, T. T. Martin, Regional Administrator, NRC Region I, said "the failure to follow operating procedures and effectively monitor and supervise plant operations resulted in the plant being placed in a condition potentially outside your design, since the SW (service water) and ESW (emergency service water) systems provide the ultimate heat sink for this unit." Martin did recognize that the "operators demonstrated good performance subsequent to the event by promptly restoring service water flow to essential components and placing the plant in cold shutdown."

Shearon Harris: Inoperable Reactor Trip System Instrumentation Channel

A civil penalty in the amount of \$50 000 was imposed on July 23, 1991, on the Carolina Power & Light Company, the licensee of the Shearon Harris nuclear plant.[28]

The Notice of Violation said the fine was being imposed because plant personnel failed to maintain two operable Reactor Trip System instrumentation channels from May 18 to June 3 and did not ensure that operations personnel were properly notified prior to placement or removal of electrical jumpers during testing of the system.

The Reactor Protection System provides for automatic reactor shutdown if instrumentation indicates certain unsafe conditions. With one channel inoperable, the system would not have been able to perform its intended safety function if a single failure involving the other channel had occurred. According to the NRC notice, company maintenance personnel had used electrical jumpers and other equipment to conduct tests. This apparently resulted in the undetected failure of a transistor, which caused the problem.

South Texas Project: Unavailable Shutdown System

The NRC imposed a fine of $75 000 against the Houston Lighting and Power Company (HL&P) because of periodic unavailability of a system that NRC requires PWR plants like South Texas Project (STP) to have.[29] This system, required by 10 CFR 50.62, is intended to deal with "Anticipated Transients Without Scram" (ATWS) events and is called the AMSAC, or ATWS Mitigating System Actuation Circuitry. In Westinghouse PWRs it provides alternate signals for auxiliary feedwater pumps and for turbine trip.[30]

Both NRC regulations and the plant safety analysis report require that the system be highly reliable and be available whenever the plant operates above 40% power. During an inspection May 16 to June 3, 1991, NRC inspectors found that, in STP 1, the system had been out of service about 36% of the time since October 1989 while the plant was above 40% power. In Unit 2, they found that it had been out of service about 15% of the time the plant was above the 40% power level since the start of commercial operation. Furthermore, the inspectors also discovered that, on May 16, 1992, the Unit 2 system would not have automatically initiated because some circuits were inactivated.

In his letter informing HL&P of the civil penalty, R. D. Martin, NRC regional administrator in Arlington, Tex., said NRC has "significant regulatory concerns" that the utility failed to assure the proper functioning and reliability of this system and failed to remain cognizant of the regulation requiring it. He said the utility had taken immediate actions and made long-term plans to improve the system's reliability. Without credit for these corrective actions, Martin's letter said, the civil penalty would have been higher.

Browns Ferry 2: Open Personnel Access Doors

A $75 000 civil penalty against the Tennessee Valley Authority (TVA) was assessed for alleged violation of NRC requirements at the Browns Ferry nuclear power plant near Athens, Ala.[31]

The NRC said the proposed fine was for an event on June 5, 1991, in which both Unit 2 drywell personnel access doors were open at the same time for a period of about 4 hours during power ascension testing, which resulted in a loss of containment integrity during power ascension tests. Apparently a mechanical maintenance employee, assigned to operate the personnel access doors for workers entering the drywell to make visual observations during the tests, disarmed the door interlocks at 2:45 a.m. and left the area. The individual informed a general foreman that the doors were open. The general foreman informed a lower level foreman, and two teams of test personnel entered and exited the drywell without expressing concern that the doors were open.

Plant Technical Specifications require that one of two doors in a personnel access airlock be closed at all times when the plant is operating. NRC officials said that Browns Ferry personnel failed to maintain containment integrity properly, that a trained and qualified maintenance mechanic failed to comply with the procedures for defeating the drywell personnel air-lock door interlock, and that TVA failed to develop and implement appropriate procedures to control nonroutine containment entry. The NRC also said it was concerned that several significant opportunities were missed to detect and correct the problem.

Vermont Yankee: False Central Alarm Station Computer Entries

The NRC staff proposes to fine Vermont Yankee Nuclear Power Corporation $75 000 for alleged safeguards violations at the Vermont Yankee plant near Brattleboro, Vt.[32] An NRC inspection was conducted at the plant on June 6 through 10, 1991, to look into the circumstances surrounding an event at the facility in May 1992, which involved the failure of the Central Alarm Station (CAS) Operator to respond to an alarm in accordance with Vermont Yankee's NRC-approved security plan. The NRC inspectors determined the CAS operator had made entries into the security computer, which indicated that he had taken the appropriate actions, but the

NRC staff alleges that the CAS operator had actually not taken those actions. In addition, two Secondary Alarm Station operators also failed to confirm that the proper actions had been taken.

In a letter to Vermont Yankee, T. T. Martin, Regional Administrator, NRC Region I, said, "These failures represent a serious breakdown in a safeguards system designed to prevent unauthorized or undetected access to the protected area, and collectively reflect a lack of attention by VY management toward licensed responsibilities." Martin went on to say, "the NRC recognizes that disciplinary action was taken against the CAS operator, including time off without pay, and the two SAS operators."

The normal fine for such a violation is \$50 000. In this case, however, it was increased by 50% to \$75 000 because NRC determined that the licensee's corrective actions were inadequate. This action was based on three violations involving the failure by Central Alarm Station and Secondary Alarm Station operators to take appropriate actions to respond to protected area intrusion alarms. No civil penalty was assessed because of a reporting violation associated with this incident. The escalation and mitigation factors were considered, and the base amount of the civil penalty was increased 50% because of the licensee's poor corrective actions.

FitzPatrick: Inadvertent Radioactivity Release from Waste Concentrator

The NRC staff has cited the New York Power Authority (NYPA) for alleged violations related to the inadvertent and unmonitored release of radioactive material from the James A. FitzPatrick Nuclear Power Plant, near Scriba, N.Y., on Mar. 18, 1991, and has proposed a \$137 500 fine.

On that date radioactive material was released from FitzPatrick's liquid waste concentrator to the atmosphere through an unmonitored vent intended for releasing normally clean steam from the plant's auxiliary boiler system. The ground next to the boiler and along both sides of the turbine, reactor, and control buildings and adjacent building walls and roofs were contaminated. Rain washed some of the contamination into the storm drain systems, which carried it offsite to Lake Ontario. According to the NRC staff, the concentration of radioactivity in the released material was 65 times as large as the limit specified in the plant's Technical Specifications, which are based on the limits set in NRC regulations.

The NRC staff found two violations of regulatory requirements associated with this incident, one of which was the release. The second was that NYPA was required to establish procedures for operation of the facility and systems to prevent an inadvertent release of radioactive material to the environment. At the time of the release, the process of draining or partial draining of the radwaste concentrator was being performed despite the lack of a written procedure describing this operation. In addition to the violations, the NRC expressed concern that, although the licensee's architect engineer determined in April 1975 that a potential unmonitored release pathway existed from the auxiliary boiler vents to the atmosphere and a 1980 NRC bulletin identified a similar situation at another facility, the condition at FitzPatrick was not corrected. Furthermore, licensee procedure reviews in 1987 and 1990 identified that the radwaste evolutions could not be performed as specified in the procedures, yet the procedures were not updated.

In a letter to NYPA, T. T. Martin, regional administrator, NRC Region I, said, "The NRC considers this event serious since it resulted in the release of radioactive materials to unrestricted areas. Although the radiological significance was limited by the fact that the release existed for only a short period, and actions in response to the event were prompt and effective in mitigating the consequences of the release, the NRC is concerned that the event occurred as the result of inadequate control of activities at the facility."

Although the civil penalty for these violations is normally \$50 000, the fine was increased in light of the licensee's poor past performance and prior notice of various problems with the radwaste system.

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HARVARD SCHOOL OF PUBLIC HEALTH ANNOUNCES SHORT COURSES

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Occupational and Environmental Radiation Protection, August 17–21, 1992. The topics in this course offering include atomic structure and radioactivity, sources and types of ionizing radiation, biological effects of exposures, external and internal radiation hazards, radiation monitoring and instrumentation, protection standards and dosimetry, and inspection and radiation guidelines.

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For additional information on these programs, please contact Mary F. McPeak, Office of Continuing Education, Harvard School of Public Health, 677 Huntington Ave., Boston, MA 02115. Tel: (617) 432-1171. FAX: (617) 432-1969.

16TH BIENNIAL ANS TOPICAL MEETING ON REACTOR OPERATING EXPERIENCE: PRESENT AND FUTURE TECHNOLOGIES—APPLYING LESSONS LEARNED (Call for Papers)

Melville, Long Island, N.Y., August 15–19, 1993

This American Nuclear Society (ANS) Topical Meeting is sponsored by the ANS Reactor Operations Division and the ANS Long Island Section. This topical symposium will provide an international forum for discussing technical and scientific issues based on the collective operating experience of both commercial power reactors and test, research, and training reactors. The theme of this meeting is the application of present and future technologies to the operation of existing reactors and to the design of new reactors. Authors are invited to submit papers on topics related to accident management; facility management; plant performance; operator training and performance; innovations in procedures; maintenance improvements; applications of probabilistic risk assessment (PRA); control room enhancements; advanced instrumentation and control; plant modifications; life extension; use of simulators; self assessment and root cause analysis; surveillance testing; testing, research, and training reactor (TRTR) activities; DOE Class A and B reactors; and new reactor design. The deadline for submission of 1000-word summaries (an original and three copies) plus figures and tables is August 1, 1992. Full manuscripts are due for peer review by January 15, 1993.

For information about this meeting, and to submit summaries, contact Technical Program Chairman Mr. Robert E. Hall, Brookhaven National Laboratory, Building 130, Upton, New York 11973. Telephone: (516) 282-2144. FAX (516) 282-3957.

Recent Developments

Edited by E. G. Silver

General Administrative Activities

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

This regular feature of *Nuclear Safety* summarizes selected current topics related to nuclear safety that do not fit elsewhere in the journal. Included in this issue are items reported during the third quarter of 1991. Subjects discussed, among others, are new NRC rules on fitness for duty and monitoring of maintenance programs, a discussion on conditions for "inspections, tests analyses, and acceptance criteria" for advanced reactor design certification, and the results of a major cancer study on shipyard workers involved with the nuclear navy.

ACRS COMMENTS ON SEVERAL ISSUES

During the third quarter of 1991, the Advisory Committee on Reactor Safeguards (ACRS) reported on a number of issues to the Chairman of the Nuclear Regulatory Commission (NRC). Three of these reports are excerpted here.

Use of PRA in the Regulatory Process

In its July 11–13, 1991, meeting, the ACRS discussed its concerns about the "unevenness and inconsistency" with which use is made of Probabilistic Risk Assessment (PRA). Although agreeing that this technique can be a "valuable tool for judging the quality of regulation, and for helping to insure the optimal use of regulatory and industry resources," the ACRS felt that they would like to see "a deeper and more deliberate integration of this method into the NRC activities." Their letter of recommendations reads, in part:[1]

> PRA is not a simple subject, so there are wide variations in the sophistication with which it is used by the various elements of NRC. There are only a few staff members expert in some of the unfamiliar disciplines—especially statistics—that go into a PRA, so it is not surprising that there are inconsistencies in the application of the methodology to regulatory problems.
>
> To illustrate the problems, let us just list a few of the fundamental aspects of the use of PRA, in which different elements of the staff seem to go their own ways. These are just illustrations, but each can lead to an erroneous regulatory decision.
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> 1. The proper use of significant figures is in principle a trivial matter, but it does provide a measure of a person's understanding of the limitations of an analysis. Yet we often hear from members of the staff who quote core-damage probabilities to three significant figures, and who appear to believe that the numbers are meaningful. It is a rare PRA in which even the first significant figure should be regarded as sufficiently accurate to play an important role in a regulatory decision, but there is something mesmerizing about numbers, which imbues them with misleading verisimilitude. They deserve respect, but not too much, and it is wrong to err in either direction.
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> 2. Closely related is uncertainty. There is no way to know how seriously to take the results of a PRA without some estimate of the uncertainty, yet we often hear thoroughly unsatisfactory answers (some perhaps invented on the spot) when we ask about uncertainty. One of the advantages of PRA is that it provides a mechanism for estimating uncertainty, uncertainty which is equally present, but not quantified, in deterministic analyses.
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> 3. Conservatism. A PRA should be done realistically. The proper time to add an appropriate measure of conservatism is when its results are used in the regulatory

[a]Oak Ridge National Laboratory.

process. If the PRA itself is done with conservative assumptions (more the rule than the exception at NRC), and is then used in a conservative regulatory decision-making process, self-deception can result, or resources can be squandered.

The inconsistent use of conservatism was illustrated by a pair of briefings at our April 1991 meeting, which included updates on proposed rules on license renewal and on maintenance. In the former case, we were told that a licensee could use PRA to add an item for later review, but never to remove one—a one-way sieve. In the latter case we were told that PRA could be used to justify either enhancement or relaxation of maintenance requirements. Foolish consistency may be a hobgoblin, as Emerson said, but there is nothing foolish in seeking consistency in regulation.

4. The bottom line. It has been widely recognized since WASH-1400 that the bottom-line probabilities (of either core melt or immediate or delayed fatalities) are among the weakest results of a PRA subject to the greatest uncertainties. (That doesn't mean they are useless, only that they should be used with caution and sophistication.) Yet we find staff members unaware of these subtleties, often dealing with small problems, justifying their actions in terms of the bottom line probabilities. This is only in part due to the Backfit Rule, which almost requires such behavior; it is also inexperience and lack of sensitivity to the limitations of the methodology.

   A number of staff actions and proposals use bottom-line results of a PRA as thresholds for decision making, often with the standard litany about the uncertainty in the reliability of these results. In fact, the quantified uncertainty in the bottom-line results of a PRA is just as important a number as the probability itself. It would be straightforward to employ a decision-making algorithm that prescribes a confidence level for the decision, and uses both the bottom-line probability and the uncertainty to achieve this. A further improvement would be to incorporate the consequences of erroneous decisions, what statisticians would call the loss function, into the decision-making process. The Commission has come close to this approach in its recent instructions to the staff on the diesel generator reliability question.

These are just a few examples of problems with the use of PRA in NRC, all common enough to be disturbing, and increasing in frequency as the use of PRA increases. It has been more than fifteen years since the publication of WASH-1400, a pioneering study which, despite known shortcomings, established the NRC at the forefront of quantitative risk assessment. One could have hoped that by now a coherent policy on the appropriate use of PRA within the agency, on both large and small problems, could have evolved.

We recommend that:

A. A mechanism be found (perhaps a retreat) through which the few PRA and statistical experts now scattered throughout the agency (and generally ignored) can be brought together with the appropriate senior managers and outside experts, to work toward a consistent position on the use of PRA at NRC. It could be worth the time expended. (Among other long-term benefits, such an interaction would add an element of horizontal structure to the NRC's predominantly vertical organization.)

B. The Commission then find a way to give credence and force to that position.

C. The Commission emphasize recruitment of larger numbers of professionals expert in PRA and statistics.

D. The Commission consider some kind of mandate that any letter, order, issue resolution, etc., that contains or depends on a statistical analysis or PRA, be reviewed by one of the expert PRA or statistical groups.

We do not pretend that this is an easy problem. The solution involves not only a cultural shift, so that those few experts already at NRC have some impact, but also substantial enhancement of the staff capabilities. That will require incentives that only the Commission can supply. It is interesting that the Commission's Severe Accident Policy Statement, dated August 1985, stated that "within 18 months of the publication of this severe accident statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs. . . ."

ACRS members Harold W. Lewis and J. Ernest Wilkins appended the following additional comments to this letter:

> We thoroughly endorse this letter, and regret only that the Committee chose to ignore the parallels between the PRA problems and those in a number of other newer technologies significant to nuclear safety. Recommendation C should have included mention of some of these—electronics and computers, for example—which are of increasing importance. Weaknesses in those areas also need correction. Computerized protection and control systems, in particular, require the kind of sophisticated review that NRC is in no position to provide.

Issues Raised Regarding GE's ABWR Design

The ACRS has had a continuing interest in the designs for advanced reactors, and, in particular, the General Electric Company's Advanced Boiling Water Reactor (ABWR), for which a design certification has been applied for in accordance with 10 CFR Part 50, Appendix O. In November 1989 the ACRS issued a letter on its evaluation of Module 1 of the Draft Safety Evaluation Report (DSER).[2] It has now again addressed this matter and issued the additional comments excerpted as follows:[3]

> Our previous letter to you concerning the ABWR design was dated November 24, 1989, and conveyed our

comments on Module 1 of the Draft Safety Evaluation Report (DSER). Since this letter, we have been kept apprised of the design and the status of the review while awaiting receipt of additional DSERs. The staff now says that DSER preparation by modules will be discontinued in favor of preparation by SSAR chapters and Standard Review Plan (SRP) sections.

To ensure the completeness of our review, it will be necessary to account for any additions or revisions to each DSER as forwarded by a SECY subsequent to issuance of our respective comment letter. An arrangement acceptable to us is needed to ensure the identification of any additions or revisions, and we should agree on an appropriate time for their review. Our comments will not be complete, however, until we have submitted a report to the Commission concerning the final SER on which we expect to comment by mid-November 1992.

Our activities subsequent to the completion of our November 1989 letter have focused on several design concerns that were discussed with GE and the NRC staff in an effort to ensure an early awareness and understanding. We believe that it is appropriate to document them here for timely consideration and resolution in appropriate DSER sections. We expect to have additional items later. We do not expect separate replies to our concerns provided the staff responds in the appropriate DSER.

1. Control Building Flooding

The proposed ABWR design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building with the essential 250-V DC battery rooms immediately above, and the main control room at the next higher elevation. This arrangement places the main control room below ground grade. Our concern with this arrangement is the potential for control building flooding due to an un-isolated break in the open-cycle cooling water piping or components inside the building. The ultimate heat sink (cooling pond) is likely to provide sufficient water to flood the building to near ground grade.

2. Physical Separation Barriers

Internal plant flooding and external events such as fire are of major concern if their effects cannot be confined to a single division of required safe-shutdown equipment. We believe that the key to confinement is the provision of an appropriate separation barrier. However, a classical barrier such as the 3-hour-rated fire barrier may not of itself, be sufficient to ensure divisional separation under the combined effects of pressure, heat, smoke, and flooding which accompany a fire and its mitigation. Also, it would appear from the SRP that the effects of delayed suppression on room temperature, pressure, and barrier leakage need to be considered when determining that safe shutdown can be achieved. We remain unconvinced that divisional separation barriers for the ABWR have been adequately prescribed for the range of events and conditions during which they must provide separation.

Of particular concern is a diesel fuel fire which may be subject to delayed suppression in the ABWR diesel generator rooms which are located inside the reactor building. It is not clear how these rooms will be qualified by design or testing to withstand burning fuel if spread across the floor by a fuel line rupture. Furthermore, it is not apparent how the compartment doors will be qualified for this condition or whether they can confine the fuel to the room. If manual mitigation is required, a fire barrier door must be opened. It is not certain that this can be achieved safely or that the external environmental effects of a prolonged opening of the door have been considered.

3. Environmental Protection for Solid-State Electronics

The ABWR makes extensive use of solid-state electronic components for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe rupture, fire, internal flooding, or loss of room cooling may create an adverse environment. The response of such components to the environmental change may be unpredictable and lead to unacceptable system interactions or responses. The behavior of solid state electronic components in environments created by off-normal or accident situations needs to be considered before the adequacy of any physical separation and environmental control measures can be evaluated.

4. Review of Chilled-Water Systems

The ABWR makes extensive use of large chilled-water systems to provide essential environmental cooling functions including those for the solid-state electronics. Since there is no SRP for chilled-water systems, the staff uses other guidance such as SRP section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when performing its safety evaluation. This guidance does not include evaluation of the large refrigeration equipment that is required for chilling the closed-cycle cooling water.

The NRC staff and GE need to evaluate the safety implications of chilled-water systems, including performance under varying accident heat loads, loss-of-offsite-power loading characteristics, and ability to restart and function after a prolonged station blackout. The NRC staff should develop appropriate guidance for such reviews by preparing a suitable SRP now.

5. Use of Leak-Before-Break Methodology Outside of Primary Containment

In our report of March 14, 1989 to then NRC Chairman Zech on "Additional Applications of Leak-Before-Break Technology," we expressed our belief that an avenue for consideration of further extension of the leak-before-break (LBB) concept should exist. This is still our position. We are concerned that the NRC staff is not giving serious consideration to GE proposals to extend the concept to systems outside of the primary containment because the staff feels constrained by General Design Criterion 4 which does not propose review of methodology.

We would like to see a renewed effort by GE and the NRC staff to determine if a real potential for substantial safety and/or economic benefits can be realized in applying properly the LBB concept outside of the primary containment.

6. Use of Integral Low-Pressure Turbine Rotors

The catastrophic failure of a low-pressure (LP) turbine rotor can lead to high-energy missiles that are capable of damaging safety-related equipment. The domestic turbine manufacturers (General Electric and Westinghouse) have been using an LP turbine design for large turbine generators consisting of a relatively small-diameter bored shaft with shrunk-on and keyway locked blade ring disks. The manufacturers are now offering an integral LP turbine rotor machined from a single large-diameter forging. A rotor of this design would operate at much higher stresses than the shaft of a shrunk-on disk rotor.

We were told by the Electric Power Research Institute (EPRI) representatives that a decision has not as yet been made with respect to a requirement in the ALWR Utility Requirements Document for boring the LP turbine rotors. Boring has historically been performed to remove impurity inclusions near the forging centerline. Such inclusions are stress risers and have led in the past to a number of catastrophic turbine and generator rotor failures in fossil-fueled power plants. Modern forging practices minimize such inclusions and present-day nondestructive examination and evaluation techniques provide much greater assurance of the soundness of turbine generator rotors.

The NRC staff should follow this issue closely since the use of integral LP turbine rotors, particularly if they are not bored, will require the development of an entirely new set of preoperational and periodic operational inspection, evaluation, and acceptance requirements to protect against turbine missiles. (The staff should also consider this issue for LP turbine rotor replacement programs for currently operating plants.)

7. Cavity-Floor Area Beneath Reactor Vessel

The layout of the containment for the proposed ABWR design makes use of a cavity floor area beneath the reactor vessel to deal with core/concrete interaction. This area is based on an EPRI requirement of 0.02 m^2 per Mwt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all Advanced Light Water Reactor designs.

Question of Requirements for Design Certifications Reviewed

On Sept. 10, 1991, the ACRS reported on its position with respect to the question of inspections, tests, analyses, and acceptance criteria (ITAAC) required for the issuance of a Final Design Approval (FDA) for standardized plants.[4] Their report reads, in part:

> The industry and NRC staff appear to have reached an agreement on the general features of ITAAC. However, there are still open questions on the scope and details of ITAAC and the role of the "validation attributes."
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> In SECY-91-210, the NRC staff requests commission guidance on an industry proposal that would allow the staff to issue final design approvals (FDAs) for standardized plants prior to staff approval of the proposed ITAAC. While the regulations require an applicant for a design certification FDA to submit proposed ITAAC, the contents of the FDA itself are not specified in 10 CFR Part 52. The staff has identified three possible policy options, including a proposed approach from NUMARC to resolve this issue. For the Advanced Boiling Water Reactor (ABWR), we were told that much work remains to complete the final ITAAC. However, a proposed ITAAC is expected to be submitted to the staff in December 1991, a year before the scheduled issuance of the FDA. Although the staff recommends Option 2, we believe that Option 3 is preferable. Option 3 would allow the staff to issue the FDAs only for the GE ABWR and the CE System 80+ before completing the ITAAC review and approval and then reevaluate the process for future applications.
>
> The adoption of Option 3 should not affect the staff's safety reviews or result in additional backfit constraints on the staff, since the Commission had previously commented in its February 15, 1991 SRM on the provisions of 10 CFR Part 52 by stating that "ITAAC are to provide reasonable assurance that a plant which references the design is built and will operate in accordance with the design certification, and thus are not to be used to reach a final conclusion on any safety question associated with the design. ITAAC should not be used to impose additional design requirements."

NUCLEAR INDUSTRY DESCRIBES ITS APPROACH TO MEETING NRC's ITAAC REQUIREMENTS

In addition to the ACRS position discussed previously, members of the nuclear industry also commented to NRC on the matter at a hearing held for the purpose.[5]

The industry representatives present were W. Rasin, vice president and director of the technical division at the Nuclear Management and Resources Council, D. Rehn, division manager of Catawba engineering at Duke Power, and J. Quirk, manager of the advanced light water reactor licensing certification programs at General Electric nuclear energy.

The industry defined ITAAC to be those criteria required by 10 CFR Part 52 and documented in Tier I of a design certification and/or combined license, which are necessary and sufficient for providing reasonable assurance that, if the analyses are performed and the acceptance criteria are met, the nuclear power facility is built and will operate in accordance with the provisions of the design certification and combined license.

Rasin stated that the key issues requiring attention on ITAAC, in the industry's view, are (1) validation attributes, role and scope; (2) appropriate ITAAC detail; and (3) the relationship of final design approval and ITAAC. Rasin said that the industry approach to ITAAC formulation is that a comprehensive set of criteria are needed to provide a basis for discussions and that "discrete" ITAAC examples are of limited use as generic guidance. Interactions with NRC staff were to continue in parallel with development of an integrated lead ITAAC submittal via GE's ABWR to be quickly followed by Combustion Engineering's System 80+ ITAAC submittal. The GE submittal was to incorporate the lessons learned from ABWR, said Rasin.

Rehn noted that the industry's basic strategy is to take a "two-tiered" approach to design certification, and said that "Tier II is already in place—the design, construction and operational phase." Rehn explained that, in the industry's view, Tier I contains descriptions of the certified design and ITAAC provisions and that Tier II contains the scope and standard analysis report design description, the generic quality assurance program, generic technical specifications, and "validation attributes."

Nuclear Regulatory Commission Chairman I. Selin asked repeated questions on the nature of the validation attributes, saying he was unclear what their purpose was and why they are included in Tier II as opposed to the first. Rehn responded first, saying that validation attributes are specified when direct verification of a Tier I ITAAC is not possible. As examples, Rehn gave seismic qualifications or a loss-of-coolant-accident scenario, saying that "certain analyses don't apply" for a Tier I ITAAC. Rehn continued that the validation attributes apply *only to verification* of Tier I requirements. Tier II requirements are to be verified by the quality assurance program and routine NRC inspection programs.

Selin stated that the Commission found it difficult to track what the industry was saying about validation attributes and voiced his concern that "NRC's silence does not mean acquiescence." He added that the Commission would wait for other concrete examples of what the industry means by "validation attributes."

Following up on the Chairman's question, Rehn and Rasin said that the attributes are in Tier II to deal with "potential variations" that might arise in the certification process. Rasin said that during testing the industry would likely have some "discomfort" in specifying pre-set values for particular tests: "there are things you don't have to agree on up front," said Rehn.

On the ITAAC–FDA relationship, Quirk noted that ITAAC are not to be used to impose additional design requirements, do not substitute for design detail, and are not to be used as a basis for a safety determination on the design. The industry position is that FDA issuance should be decoupled from ITAAC approval. The FDA should demonstrate accomplishment of a key NRC interim milestone, should enhance confidence in the new licensing process, and should require individual design certification applicants to assess risks.

Commissioner J. R. Curtiss disagreed and stated his belief that the design decision should be made in the FDA. "There is merit in looking at ITAAC prior to issuing FDA," said Curtiss. Curtiss noted that he disagreed with Selin in that he is "not comfortable" with the Chairman's idea that an FDA could be issued if an ITAAC is held up. Curtiss said he thinks that nullifies the finality of the design analysis; hence he would rather label an FDA at that point as "preliminary."

After the industry presentation NRC staff, headed by T. Murley of the Office of Nuclear Regulatory Research, requested Commission guidance on the form and content of ITAAC for a design certification rule and a combined license as required by 10 CFR Part 52. The staff's latest paper describes how the ITAAC for design certification, the ITAAC associated with site-specific design information, and the Tier II validation attributes constitute a verification program that can be implemented by the combined license holder. Murley said that the paper also describes how the successful completion of the ITAAC requirements and any other acceptance criteria in the combined license will constitute the basis for the NRC's determination to allow operation of the facility.

In discussing the content of the new staff paper, Murley noted that ITAAC for a combined license application are set forth in section 52.79(c) and the ITAAC for a design certification application are set forth in section 52.47(a)-(1)(vi). If the applicant for a combined license references a certified design, then the application must adopt the ITAAC for the certified portion of the design.

Murley also stated the following:

> Although the two sections of the rule cited above are applicable to different stages in the Part 52 process, their basic intent is the same. Both sections require that the applicant propose a set of ITAAC (verification activities) that will demonstrate that the facility has been properly constructed in accordance with the design and will operate in accordance with the design and will operate in conformity with applicable requirements. The use of verification activi-

ties is not unique to the Part 52 licensing process. In the 10 CFR Part 50 licensing process, the applicant (and later the license holder) was required by regulation, license condition, and final safety analysis report commitments to perform a wide range of tests and inspections before the NRC issued a full power license. What is new, however, is that under the Part 52 process these verification activities and their associated acceptance criteria will be specified in the design certification rule, and later the combined license. The benefits to the early designation of these verification requirements include an up-front agreement to requirements and acceptance criteria, and the consolidation of requirements into a single document prior to commencement of construction activities.

NRC MOVES TO RATIONALIZE SAFETY DEFECTS REPORTING

In July 1992 the NRC amended Parts 21 and 50 of 10 CFR, with an effectiveness date of Oct. 29, 1992, with a view to making the reporting of safety defects found in the course of design, construction, and operation of nuclear facilities more rational and useful.[6]

Under various portions of the Commission's regulations, operators of nuclear power plants and non-power reactors, holders of construction permits, and non-licensees supplying basic components for NRC-licensed facilities or activities are all required to report safety defects to various parts of the NRC staff. The amendments to these regulations will, among other things:

—Eliminate instances, which have occurred in the past, of more than one organization evaluating and reporting a safety defect as the result of having to satisfy different parts of the NRC regulations.

—Establish uniform time limits for initial reports of safety defects (within 2 days of the determination that a safety defect exists) and for follow-up reports (within 30 days of the determination).

—Establish a 5-day limit for vendors to transfer information to end users when it is not possible to determine whether a safety defect in fact exists.

—Establish a uniform definition of defects to be reported.

—Establish a requirement to ensure the uniformity of all safety defect reports.

—Clarify the term "basic component."

—Specify time limits for retention of specific records.

—Provide that, in cases where an evaluation of a potential safety defect cannot be made within 60 days, an interim report be submitted within 60 days of discovery of the potential safety defect.

NRC AMENDS REGULATIONS TO REQUIRE EMERGENCY RESPONSE DATA SYSTEM

In our previous issue[7] we published a report on the ACRS views of the proposed Emergency Response Data System (ERDS) for nuclear power plants. The requirement for such a system has now been issued as an NRC Regulation that requires licensed nuclear power plants to participate in the ERDS. The rule applies to all operating reactor power reactor facilities except Big Rock Point (which is exempt because the plant configuration does not permit collection of sufficient data to effectively participate in ERDS) and those which are permanently or indefinitely shut down.

The ERDS is a direct electronic data link between computer data systems used by licensed utilities and the NRC's Operation Center and supplements the voice transmission of information over the currently installed Emergency Notification System. It is to be activated by a licensee when an alert or higher-level emergency event occurs at a nuclear power plant.

The NRC stated that it needs this system to carry out its role in the event of a nuclear power-plant emergency, which is to monitor licensee actions to ensure that recommendations are made with respect to off-site protective measures. In addition, the NRC expects to provide technical analysis and logistical support to the licensee, support off-site authorities such as state and local governments, keep other federal agencies informed of the status of the emergency, keep the media informed of the NRC's knowledge of the status of the emergency, and coordinate with other public affairs groups. The voice-only system, which was placed in service shortly after the 1979 Three Mile Island accident, requires excessive amounts of time for routine transmission of data and for data verification or correction. In addition, errors have been attributed to the transcription and interpretation of voice-transmitted data.

The new rule requires utilities to provide the necessary computer software to assemble the data and an output communication port for each reactor unit in its on-site computer system. The required data on plant conditions will be transmitted to the NRC Operations Center (NRCOC) in Bethesda via NRC-provided communication link hardware. The system is to be activated in the event of any alert, site area emergency or general emergency. Licensees will be required to have the system operable within 18 months of the effective date of this rule (Sept. 12, 1991) or before initial escalation to full power, whichever comes later.

The ERDS began as a voluntary program, dating back to August 1989, when the NRC staff first requested voluntary participation in the ERDS program. Currently, about half the licensed nuclear power plants have volunteered to participate in the ERDS, and about ten reactor units already are capable of transmitting ERDS data to the NRCOC. The new rule is intended to ensure expeditious implementation of the ERDS program at all nuclear plants.

REQUIREMENTS FOR QUALITY MANAGEMENT PROGRAMS FOR ISOTOPE USERS ISSUED

The NRC-licensed users of radioactive materials for therapeutic procedures, as well as certain other users of radioiodine, will be required to implement quality management programs to ensure that the materials and irradiations are used as specified by the prescribing physicians.[8] About 6 000 license holders from either the NRC or so-called "Agreement States" will be affected.

An estimated 30 000 therapeutic procedures are performed each year with radiopharmaceuticals. In these procedures radioactive drugs are administered to treat hyperactive thyroid conditions and certain forms of cancer.

Sealed radioactive sources that produce high radiation are also used to treat cancer. About 100 000 patients per year receive treatments involving the application of a beam of radiation from cobalt-60 to the part of the patient's body to be treated.

Smaller sealed sources with less radioactivity are designed to be implanted directly into a tumor area or applied on the surface of areas to be treated in a procedure known as brachytherapy. Licensees perform approximately 50 000 brachytherapy treatments annually.

The new regulation is intended to enhance patient safety in a cost-effective manner while allowing the flexibility necessary to minimize intrusion into medical judgments.

The revisions require affected licensees to implement a written quality management program that includes annual reviews and evaluations to determine whether the program is still effective. The program will have to include written policies and procedures to ensure, for example, that the patient's identity is verified by more than one method prior to administering the radioactive material or radiation.

Currently, medical licensees are required to keep a record of each misadministration of radioactive materials and, in certain circumstances, to report the mistake to the NRC. The new regulations strengthen the requirements by adding additional types of mistakes to the list of those for which record-keeping and reporting requirements apply.

These new amendments to Parts 2 and 35 of the Commission's regulations (10 CFR) became effective in January 1992 and the affected licensees, by then, had to have implemented a quality management program and submitted a copy of the program to the NRC.

GAO EXAMINES INPO INPUT TO NRC ACTIONS

A Government Accounting Office (GAO) report, based on an investigation requested by U.S. Representatives P. H. Kostmeyer (D-Pa.) and E. J. Markey (D-Mass.), noted that the NRC, on 12 different occasions over the past 2 years, had refrained from issuing information notices on safety issues because the Institute for Nuclear Power Operations (INPO) had already notified the industry about the problems.[9] The concern expressed by the requesters was that NRC may have been substituting INPO actions for its own regulatory responsibilities.

The INPO was formed by the nuclear utility industry in the aftermath of Three Mile Island in 1979 for the purpose of assisting utilities in improving the safety of plant operations. Periodic evaluations by INPO of nuclear power plant performance and operating safety determine how well INPO's performance objectives and criteria, which are broader and seek a level of performance higher than the minimum level required by NRC, are being met.

The GAO report noted that NRC and INPO had written several memoranda of agreement to enhance cooperation between the two entities. The report concluded that "NRC does not routinely use INPO evaluation reports in lieu of its own inspections to carry out its oversight of the nuclear power industry." The report noted, however, that NRC staff stated that they might use INPO reports "in rare instances." Although the GAO report stated that "our review of NRC's files of INPO documents and interviews ... did not disclose any evidence that NRC currently relies on INPO evaluations in lieu of conducting its own inspections," the report also said that NRC staff had not, in "about twelve" instances, issued its own information notice (IN) because INPO had already alerted the industry to a potential problem. The significant point is that INPO's reports are not made public, whereas NRC's INs are legally required to be so. According to the NRC, on the twelve occasions in question, INs were not published to avoid "duplication." The GAO report found that "under certain circumstances NRC has decided not to issue some information notices if INPO has already alerted industry to potential problems." The GAO report continues to note: "However, INPO's reports to industry are not publicly available. Therefore, although industry has been alerted to potential problems, the fact that a certain condition, event, or circumstance may have generic applicability to the safety of nuclear power plants is not publicly disclosed."

As an example, the GAO report cites a February 1990 instance where NRC planned to issue an information notice to point out recent problems associated with a certain type of pressure-relief valve. Although staff were preparing to draft an information notice, INPO issued its own report alerting the industry. NRC subsequently decided not to issue their public information notice since staff considered the industry sufficiently warned.

The INPO has specifically requested NRC not to publish INPO reports, and NRC has obliged. The GAO report recommends that in the future the NRC should issue its INs "without regard to whether they contain the same or similar information as INPO's communications," to guarantee public access to all information about potential safety problems.

The position of INPO of not publishing their reports is based on their view that wider distribution of the reports would tend to decrease the nuclear utilities' voluntary cooperation with INPO, hamper INPO's effectiveness, and detract from industry efforts to strive for excellence.

NRC ISSUES FINAL RULES TO CLARIFY FFD RELATIONSHIP TO DRUG AND ALCOHOL ABUSE PROBLEMS

The NRC recently took several related actions to clarify its position regarding "fitness-for-duty" (FFD) programs for licensed operators of nuclear reactors. In July 1991 it adopted a regulation that specifies that the conditions and drug testing cutoff levels established under the FFD programs are applicable to licensed reactor operators as conditions of their licenses. The rule provides a basis for taking action against licensed operators who (1) use drugs or alcohol in a manner that exceeds levels described in the FFD rule, (2) are determined by a facility medical review officer to be under the influence of any prescription or over-the-counter drug that could adversely affect their ability to safely and competently perform licensed duties, or (3) sell, use, or possess illegal drugs.[10]

The NRC asserted that the rule "will ensure a safe operational environment for the performance of all licensed activities by providing a clear understanding of the severity of violating requirements governing drug and alcohol use, and substance abuse."

To justify the medical basis for the rule, NRC carefully spelled out why operators should be required to comply: "given the nature of certain drugs, even though the presence of drug metabolites does not necessarily relate directly to a current impaired state, the presence of drug metabolites in an individual's system strongly suggests the likelihood of past, present, or future impairment affecting job activities." Accordingly, the rule requires power reactor licensees to have written policies and procedures that address FFD requirements. Besides monitoring potential abuses of prescription and over-the-counter drugs, FFD mandates that other factors, such as mental stress, fatigue, and illness, also be watched closely.

Although the Commission made clear the high level of importance it attaches to FFD for operators, it reserved for itself the right to review individual licensee enforcement actions. The Commission also noted that the revised rule does not apply to non-power facility licensees.

If an operator tests positive for drugs and exceeds the cutoff levels of 10 CFR Part 26 or the facility's cutoff levels, if lower, the following actions may ensue: (1) For the first incident, "normally only a notice of violation will be issued," barring aggravating circumstances, such as performance errors; (2) for the second incident, the operator's license could be suspended for up to 3 years, (3) the third time an operator is found to have exceeded cutoff levels, NRC states it "intends" to revoke the license.

During the public comment period on the proposed new rule, 39 respondents submitted opinions, an overwhelming majority of whom opined that the rule was unnecessary since regulations were already in place to ensure that reactor operators adhere to 10 CFR Part 26. The NRC acknowledges this fact but believes that the licensed operator is one of the main components and possibly the most critical component of continued safe reactor operation. Therefore it wants to emphasize and clearly inform the operators that, as conditions of their licenses, they must comply with their facility's FFD program. The rule became effective Aug. 14, 1991.

Then, in September 1991 the NRC issued an amendment to the FFD rule to clarify its position on the application of its new rule.[11]

The amendment clarifies the Commission's intent concerning the unacceptability of taking action against an individual on the basis solely of the preliminary, unconfirmed results of a drug screening test. It also permits, under certain conditions, temporary administrative actions, up to temporary removal of an individual from unescorted access or from normal duties, on the basis of an unconfirmed positive result from an initial screening test for marijuana or cocaine.

To minimize the impact of such administrative actions on individuals whose on-site test is not subsequently confirmed, the Commission is requiring that the testing protocols and controls provide high levels of accuracy and reliability, that there be no loss of compensation or benefits pending completion of the testing process, and that there be no disclosure or record of any suspension on the basis of a test that is not subsequently confirmed.

The decision to clarify the July 1991 FFD requirements came after one licensee advised the NRC that it had implemented an FFD program that included a provision for placing individuals in a non-work pay status on the basis of a positive but unconfirmed initial drug test. The amendment became effective on Sept. 23, 1991.

NRC ISSUES RULE ON MONITORING OF NUCLEAR POWER PLANT MAINTENANCE PROGRAMS

The NRC issued an amendment to 10 CFR Part 50 to require nuclear plant licensees to monitor the effectiveness of maintenance activities for safety-significant plant equipment to minimize the likelihood of failures and other events caused by the lack of effective maintenance.[12] (The ACRS discussion of this issue was reported in our previous issue.[13])

The new requirements result from the Commission's conclusion that "proper maintenance is essential to the safety of nuclear power plants" and on its assessment of criteria proposed by its staff to be used in determining the need for a maintenance rule. As a result of this assessment, the Commission found that

—Maintenance team inspections by NRC staff have shown that licensees have adequate maintenance programs and have exhibited an improving trend in implementing those programs. However, some common maintenance-related weaknesses, such as inadequate root cause analysis leading to repetitive failures, lack of equipment performance trending, and the consideration of plant risk in the prioritization, planning, and scheduling of maintenance, were identified.

—The industry, through the Nuclear Utilities Management and Resources Committee, expressed a general commitment to improving performance in the maintenance area. However, no written commitments were received from individual licensees, and the Commission believes that the indirect, industry-wide commitment does not constitute a sufficient commitment from licensees in this regard.

—The industry indicated that all licensees will perform comprehensive assessments of individual licensee maintenance programs on a one-time basis over a period of 4 years against performance objectives established by INPO. In addition, INPO will continue to conduct periodic evaluations of nuclear power plant performance, including maintenance. However, the Commission believes these efforts, which largely are programmatic assessments and evaluations, will not alone suffice since the effectiveness of maintenance programs must be assessed on an ongoing basis to ensure that key structures, systems, and components are capable of performing their intended function and that there is feedback of the results of such assessments.

Accordingly, each utility licensed to operate a nuclear power plant will be required to

—Monitor the performance or condition of structures, systems, or components against licensee-established goals that are designed to provide reasonable assurance that they are capable of fulfilling their intended functions. These goals are to be established commensurate with safety and, where practical, take into account industry-wide operating experience. When a structure, system, or component does not meet established goals, appropriate corrective action is to be taken. A monitoring program will not be required in instances where it can be demonstrated that the performance or condition of a structure, system, or component is being effectively controlled and remains capable of performing its intended function through a preventive maintenance program.

—Evaluate, at least annually, the performance and condition monitoring activities and associated goals, taking into account, where practical, industry-wide operating experience, and adjust as necessary. Adjustments are to take into account the objective of preventing failures while minimizing the unavailability of structures, systems, and components as the result of monitoring or preventive maintenance. In addition, in performing monitoring and preventive maintenance activities, an assessment is to be made of the total plant equipment that is out of service to determine the overall effect on the performance of safety functions.

—Include in the scope of the monitoring program all safety-related structures, systems, and components that are relied on to remain functional during and following design-basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe, shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite radiation doses comparable to guidelines specified in the NRC's regulations.

—Also include those non-safety-related structures, systems, and components that are relied on to mitigate accidents or transients (off-normal operating events) or are used in plant emergency operating procedures, or whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function, or whose failure could cause a reactor scram or actuation of a safety-related system. This amendment is to become effective in July 1996.

DOE's SHIPYARD RADIATION STUDY FOUND NO RADIATION CANCER BUT FINDS EVIDENCE FOR POSSIBLE ASBESTOS-RELATED MESOTHELIOMA

The results of a study conducted with DOE funding by the Johns Hopkins University on cancer risk associated with radiation exposure for shipyard workers at facilities that service nuclear-powered ships was released by DOE.[14] The study showed no cancer risks linked to radiation exposures for this worker group but did find an increase in mesothelioma, a kind of respiratory system cancer normally associated with asbestos exposure. The study covered the period from 1957 (beginning with the initial overhaul of the first nuclear-powered submarine, the *USS Nautilus*) through 1981.

The project was carried out to determine whether shipyard workers who receive some occupational radiation exposure are subject to any excess risk of cancer or other disease. It surveyed a population of more than 70 000 civilian workers from two private and six U.S. Navy shipyards.

The DOE called the study "a milestone among studies of groups of workers exposed to radiation" and claimed that it was the largest study of its kind ever conducted. The report's main conclusion was that the overall death rate (not only that caused by cancer) for the general U.S. population was higher than the death rate among radiation-exposed shipyard workers. But DOE also pointed out that many worker populations have lower mortality rates than the general population because they must be healthy to be hired and their health is monitored in the course of their continued employment. The group surveyed falls into that category.

Specifically, the study showed that the death rate for cancer among 38 220 radiation-exposed workers was slightly lower than that of the U.S. population but said that the difference was not statistically consequential. The corresponding death rate among the 32 510 non-radiation exposed workers was slightly higher (12%) because of a small increase of lung cancer. A slightly lower-than-expected rate of leukemia was found among both the radiation-exposed and non-exposed shipyard workers. But the study also found that the lowest leukemia rates were found among the radiation-exposed group. Again, DOE noted that the differences were not statistically significant.

The report noted an increased rate of mesothelioma, a respiratory system cancer related to asbestos exposure, in both the radiation-exposed and non-radiation-exposed groups. Twenty-six cases were observed in the group exposed to radiation, where the expected number would have been only five, and ten cases were observed in the non-exposed group, where only four cases would have been expected. DOE stated that "the actual number of mesothelioma cases was small, which reflects the rarity of this disease in the general population."

Rather than link the increase in mesothelioma cases to radiation exposure, DOE stated it "suspects" that the disease should be attributed to shipyard worker exposure to asbestos in the early years before the hazards associated with asbestos were well understood. DOE said that this hypothesis could help explain the increase in lung cancers and cases of mesothelioma. Though the study did not look specifically at asbestos or other possible risk factors, such as smoking, additional studies are planned to investigate the observations and update the study with data that go beyond 1981.

The principal investigator of the study was Dr. G. Matanoski, a professor of epidemiology at the Johns Hopkins University School of Hygiene and Public Health. Peer review and technical oversight were provided by a panel of external medical consultants and experts.

REFERENCES

1. Letter to I. Selin, Chairman, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *The Consistent Use of Probabilistic Risk Assessment,* cited in NRC Press Release 91-87, July 25, 1991.
2. Comments on Draft Safety Evaluation Report for Advanced Boiling Water Reactor, *Nucl. Saf.,* 31(2): 270-272 (April–June 1990).

3. Letter to J. M. Taylor, Executive Director for Operations, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *Concerns Related to the General Electric Advanced Boiling Water Reactor Design*, cited in NRC Press Release 91-87, July 25, 1991.
4. Letter to I. Selin, Chairman, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certification*, cited in NRC Press Release 91-119, Sept. 24, 1991.
5. *At. Energy Clearing House*, 37(34): 2 (Aug. 23, 1991).
6. NRC Press Release 91-90, July 31, 1991.
7. Final Rulemaking to Implement the Emergency Response Data System, *Nucl. Saf.*, 32(4): 602 (October–December 1991).
8. NRC Press Release 91-78, July 11, 1991.
9. *At. Energy Clearing House*, 27(27): 1 (July 5, 1991).
10. *At. Energy Clearing House*, 37(29): 3 (July 19, 1991).
11. *At. Energy Clearing House*, 37(36): 8 (Sept. 6, 1991).
12. NRC Press Release 91-79, July 11, 1991.
13. Staff Evaluation and Recommendations on Maintenance Rulemaking, *Nucl. Saf.*, 32(4): 604 (October–December 1991).
14. *At. Energy Clearing House*, 37(39): 4 (Sept. 27, 1991).

Reports, Standards, and Safety Guides

By D. S. Queener[a]

This article contains four lists of various documents relevant to nuclear safety as compiled by the editor. These lists are: (1) reactor operations-related reports of U.S. origin, (2) other books and reports, (3) regulatory guides, and (4) nuclear standards. Each list contains the documents in its category which were published (or became available) during the three-month period (July, August, and September 1991) covered by this issue of *Nuclear Safety*. The availability and cost of the documents are noted in most instances.

OPERATIONS REPORTS

This category is listed separately because of the increasing interest in the safety implications of information obtainable 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, 2120 L Street, NW, Washington, DC 20555.

NRC Office of Nuclear Reactor Regulation

The NRC Office of Nuclear Reactor Regulation (NRR) issues reports regarding operating experience 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 from the licensee. The NRR also periodically issues Generic Letters (GL) to licensees, usually for information purposes only.

NRC Information Notices

NRC IN 85-18, Supplement 1 *Failures of Undervoltage Output Circuit Boards in the Westinghouse-Designed Solid State Protection System*, September 10, 1991, 4 pages plus one-page attachment.

NRC IN 89-56, Supplement 2 *Questionable Certification of Material Supplied to the Defense Department by Nuclear Suppliers*, July 19, 1991, 3 pages plus 8 pages of attachments.

NRC IN 91-43 *Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate*, July 5, 1991, 4 pages plus one-page attachment.

NRC IN 91-44 *Improper Control of Chemicals in Nuclear Fuel Fabrication*, July 8, 1991, 3 pages plus one-page attachment.

NRC IN 91-45 *Possible Malfunction of Westinghouse ARD, BFD, and NBFD Relays, and A200 DC and DPC 250 Magnetic Contactors*, July 5, 1991.

NRC IN 91-46 *Degradation of Emergency Diesel Generator Fuel Oil Delivery Systems*, July 18, 1991, 5 pages plus one-page attachment.

NRC IN 91-47 *Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test*, August 6, 1991.

[a]Oak Ridge National Laboratory.

NRC IN 91-48 *False Certificates of Conformance Provided by Westinghouse Electric Supply Company for Refurbished Commercial-Grade Circuit Breakers,* August 9, 1991, 2 pages.

NRC IN 91-49 *Enforcement of Safety Requirements for Radiographers,* August 5, 1991, 3 pages plus 7 pages of attachments.

NRC IN 91-50 *A Review of Water Hammer Events After 1985,* August 20, 1991, 4 pages plus one-page attachment.

NRC IN 91-51 *Inadequate Fuse Control Programs,* August 20, 1991, 3 pages plus 2 pages of attachments.

NRC IN 91-52 *Nonconservative Errors in Overtemperature Delta-Temperature (OTΔT) Setpoint Caused by Improper Gain Settings,* August 29, 1991.

NRC IN 91-53 *Failure of Remote Shutdown System Instrumentation Because of Incorrectly Installed Components,* September 4, 1991.

NRC IN 91-54 *Foreign Experience Regarding Boron Dilution,* September 6, 1991.

NRC IN 91-55 *Failures Caused by an Improperly Adjusted Test Link in 4.16 KV General Electric Switchgear,* September 16, 1991, 2 pages plus 5 pages of attachments.

NRC IN 91-56 *Potential Radioactive Leakage to Tank Vented to Atmosphere,* September 19, 1991, 3 pages plus 3 pages of attachments.

NRC IN 91-57 *Operational Experience on Bus Transfers,* September 19, 1991, 2 pages plus one-page attachment.

NRC IN 91-58 *Dependency of Offset Disc Butterfly Valve's Operation on Orientation with Respect to Flow,* September 20, 1991, 3 pages plus 2 pages of attachments.

NRC IN 91-59 *Problems with Access Authorization Programs,* September 23, 1991, 4 pages plus one-page attachment.

NRC IN 91-60 *False Alarms of Alarm Ratemeters Because of Radiofrequency Interference,* September 24, 1991, 3 pages plus one-page attachment.

NRC IN 91-61 *Preliminary Results of Validation Testing of Motor-Operated Valve Diagnostic Equipment,* September 30, 1991, 3 pages plus one-page attachment.

NRC IN 91 62 *Diesel Engine Damage Caused by Hydraulic Lockup Resulting From Fluid Leakage Into Cylinders,* September 30, 1991, 3 pages.

NRC Generic Letters

NRC GL 91-10 *Explosives Searches at Protected Area Portals,* July 8, 1991, one page plus 2 pages of attachments.

NRC GL 91-11 *Resolution of Generic Issues 48, "LCOs for Class 1E Vital Instrument Buses," and 49, "Interlocks and LCOs for Class 1E Tie Breakers" Pursuant to 10CFR50.54(f),* July 18, 1991.

NRC GL 91-12 *Operating Licensing National Examination Schedule,* August 27, 1991, 2 pages plus 6 pages of attachments.

NRC GL 91-13 *Request for Information Related to the Resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," Pursuant to 10CFR50.54(f),* September 19, 1991, 4 pages plus 13 pages of attachments.

NRC GL 91-14 *Emergency Telecommunications,* September 23, 1991, 2 pages plus 7 pages of attachments.

NRC GL 91-15 *Operating Experience Feedback Report, Solenoid-Operated Valve Problems at U.S. Reactor,* September 23, 1991, 2 pages plus one-page attachment.

Other Operations Reports

These are other reports issued by various organizations in the United States dealing with power-reactor operations activities. Most NRC publications (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 obtained from the NRC Public Document Room (PDR). Specify the report number when ordering. Telephone orders can be made by calling (202) 634-3273.

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 (OSTI), P.O. Box 62, Oak Ridge, TN 37831. Reports available through one or more of these organizations are designated with the appropriate information (i.e., GPO, PDR, NTIS, and OSTI) in parentheses at the end of the listing, followed by the price, when available.

NUREG-0090, Vol. 14, No. 2 *Report to Congress on Abnormal Occurrences April–June 1991,* September 1991, 23 pages (GPO).

NUREG-0713, Vol. 10 *Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 1988, Twenty First Annual Report,* C. T. Raddatz and D. Hagemeyer, July 1991 (GPO).

NUREG-1445 *Regulatory Analysis for the Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,* T. Y. Chang, September 1991, 105 pages (GPO).

NUREG-1450 *Potential Criticality Accident at the General Electric Nuclear Fuel and Component Manufacturing Facility, May 29, 1991,* August 1991 (GPO).

NUREG/CR-2000, Vol. 10, No. 6 *Licensee Event Report (LER) Compilation for Month of June 1991,* July 1991, 64 pages (GPO).

NUREG/CR-2000, Vol. 10, No. 7 *Licensee Event Report (LER) Compilation for Month of July 1991,* August 1991, 70 pages (GPO).

NUREG/CR-2000, Vol. 10, No. 8 *Licensee Event Report (LER) Compilation for Month of August 1991,* September 1991, 84 pages (GPO).

NUREG/CR-4427 *Auxiliary Feedwater System Risk-Based Inspection Guide for the Byron and Braidwood Nuclear Power Plants,* N. E. Moffitt et al., Pacific Northwest Lab., Wash., July 1991, 27 pages (GPO).

NUREG/CR-5761 *Auxiliary Feedwater System Risk-Based Inspection Guide for the Salem Nuclear Power Plant,* R. Pugh et al., Pacific Northwest Lab., Wash., August 1991, 26 pages (GPO).

NUREG/CR-5763 *Auxiliary Feedwater System Risk-Based Inspection Guide for the Callaway Nuclear Power Plant,* N. E. Moffitt et al., Pacific Northwest Lab., Wash., August 1991, 26 pages (GPO).

NUREG/CR-5764 *Auxiliary Feedwater System Risk-Based Inspection Guide for the Ginna Nuclear Power Plant,* R. Pugh et al., Pacific Northwest Lab., Wash., September 1991, 28 pages (GPO).

DOE- and NRC-Related Items

NUREG-0933 *A Prioritization of Generic Safety Issues,* J. Pittman et al., July 1991 (GPO).

NUREG-1423, Vol. 2 *A Compilation of Reports of The Advisory Committee on Nuclear Waste, July 1990–June 1991,* August 1991, 99 pages (GPO).

NUREG/CR-5606 *A Review of the South Texas Project Probabilistic Safety Analysis for Accident Frequency Estimates and Containment Binning,* T. A. Wheeler et al., Sandia National Labs., N.Mex., August 1991, 332 pages (GPO).

NUREG/CR-5634 *Identification and Assessment of Containment and Release Management Strategies for a BWR Mark I Containment,* C. C. Lin and J. R. Lehner, Brookhaven National Lab., N.Y., September 1991 (GPO).

NUREG/CR-5641 *Study of Operational Risk-Based Configuration Control,* P. K. Samanta et al., Brookhaven National Lab., N.Y., August 1991, 147 pages (GPO).

NUREG/CR-5768 *Ice-Condenser Aerosol Tests,* M. L. Ligotke et al., Pacific Northwest Lab., Wash., September 1991 (GPO).

NUREG/CR-5771 *Probability and Consequences of Misloading Fuel in a PWR,* D. J. Diamond et al., Brookhaven National Lab., N.Y., August 1991, 61 pages (GPO).

NUREG/CR-5784 *Fitness for Duty in the Nuclear Power Industry. A Review of the First Year of Program Performance and an Update of the Technical Issues,* N. Durbin et al., Pacific Northwest Lab., Wash., September 1991, 80 pages (GPO).

Other Items

NSAC-137 *Maintaining Operability of Nuclear Plant Instrument Air System,* M. W. Akhtar et al., EPRI Nuclear Safety Analysis Center, September 1991 [EPRI Research Reports Center (RRC), Box 50490, Palo Alto, CA 94303].

NSAC-154 *ISLOCA Evaluation Guidelines,* E. T. Burns et al., EPRI, September 1991 (RRC).

NSAC-155 *Interfacing System Isolation Experience Review,* V. M. Andersen et al., EPRI, August 1991, 190 pages (RRC).

NSAC-156 *Residual Heat Removal Experience Review and Safety Analysis, Pressurized Water Reactors, 1982–1989,* H. R. Booth, EPRI, August 1991, 181 pages (RRC).

NSAC-171 *The Effects of Sodium Pentaborate Injection at BWRs,* J. E. Oesterle et al., EPRI, September 1991, 40 pages (RRC).

U.S. Steam–Electric Plants, Ten Year Production Costs: 1981–1990, Utility Data Institute Inc. (UDI), Washington, D.C., 1991 (UDI, 1700 K St., NW, Suite 400, Washington, DC 20006. Phone 1-800-486-3660.)

ORAU 91/J-20 *Ionizing Radiation Risk Assessment, BEIR IV,* Office of Science and Technology Policy, Washington, D.C., October 1991, 30 pp. (NTIS).

Licensing Systems and Inspection of Nuclear Installations, 1991, OECD Nuclear Energy Agency, 1991, 144 pages (OECD Publications and Information Center, 2001 L St., NW, Suite 700, Washington, DC 20036-4095).

Small and Medium Reactors, OECD Nuclear Energy Agency, 1991, 129 pages (OECD).

REGULATORY GUIDES

To expedite the role and function of the NRC, its Office of Nuclear Regulatory Research prepares and maintains a file of Regulatory Guides that define much of the basis for the licensing of nuclear facilities. These Regulatory Guides are divided into 10 divisions as shown in Table 1.

Single copies of the draft guides may be obtained from NRC Distribution Section, Division of Information Support Services, Washington, DC 20555. Draft guides are issued free (for comment), and licensees receive both draft and final copies free; others can purchase single

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

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, 2120 L Street, NW, Washington, DC 20555, 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 July, August, and September 1991 reporting period, are listed below.

Division 8 Occupational Health Guides

8.25 (Proposed Rev. 1) *Air Sampling in the Workplace,* September 1991.

NUCLEAR STANDARDS

Standards pertaining to nuclear materials and facilities are prepared by many technical societies and organizations in the United States, including the Department of Energy (DOE) (NE Standards). When standards prepared by a technical society are submitted to the American National Standards Institute (ANSI) for consideration as an American National Standard, they are assigned ANSI standard numbers, although they may also contain the identification of the originating organization and be sold by that organization as well as by ANSI. We have undertaken to list here the most significant nuclear standards actions taken by organizations during July, August, and September 1991. Actions listed include issuance for comments, approval by the ANSI Board of Standards Review (ANSI-BSR), and publication of the approved standard. Persons interested in obtaining copies of the standards should write to the issuing organizations.

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 (KTA). The KTA, formerly a component of the Gesellschaft für Reaktorsicherheit (GRS), is now integrated in the Federal Office for Radiation Protection (Bundesamt für Strahlenschutz BfS) in Salzgitter, Germany. Copies of these standards can be ordered from Dr. T. Kalinowski, KTA-Geschäftsstelle, Postfach 10 01 49, 3320 Salzgitter 1, Germany. These standards are in German and, unless otherwise noted, an English translation is available from the KTA.

Prices for the standards are shown in German currency (DM). The KTA standards are listed in this issue.

KTA

KTA 1201 *Requirements for the Operating Manual (previous version 3/81),* December 1985.

KTA 1202 *Requirements for the Testing Manual,* June 1984.

Proposed Rule Changes as of Sept. 30, 1991[a,b]

(Changes Since the Previous Issue of *Nuclear Safety* Are Indicated by Shaded Areas)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 1	12-12-88	1-30-89		Policy statement on exemptions from regulatory control	Advanced notice of proposed policy statement in 53:238 (49886)
10 CFR 2	4-3-90	6-18-90		Revisions to procedures to issue orders	Published for comment in 55:64 (12370)
10 CFR 2	7-5-90	9-4-90		Revisions to procedures to issue orders: challenges to orders that are made immediately effective	Published for comment in 55:129 (27645)
10 CFR 2 10 CFR 50 10 CFR 54	7-17-90	10-15-90		Nuclear power plant license renewal	Published for comment in 55:137 (29043); request for extension of comment period denied in 55:166 (34939)
10 CFR 2	10-24-90	12-10-90		Options and procedures for direct Commission review of licensing board decisions	Published for comment in 55:206 (42947)
10 CFR 2 10 CFR 40 10 CFR 70 10 CFR 74	12-17-90	3-4-91		Material control and accounting requirements for uranium enrichment facilities producing special nuclear material of low strategic significance	Published for comment in 55:242 (51726)
10 CFR 2 10 CFR 55			7-15-91; 8-14-91	Operators' licenses	Final rule in 56:135 (32066)
10 CFR 2 10 CFR 35			7-25-91; 1-27-92	Quality management program and misadministrations	Final rule in 56:143 (34104)
10 CFR 2	8-2-91	9-3-91	8-2-91; 8-2-91	Policy and procedure for enforcement actions policy statement	Policy statement modification in 56:149 (36998)
10 CFR 2 10 CFR 30 10 CFR 40 10 CFR 50 10 CFR 60 10 CFR 61 10 CFR 70 10 CFR 72 10 CFR 110 10 CFR 150			8-15-91; 9-16-91	Revisions to procedures to issue orders; deliberate misconduct by unlicensed persons	Final rule in 56:158 (40664)

(Table continues on the next page.)

Proposed Rule Changes as of Sept. 30, 1991 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 2 10 CFR 40 10 CFR 50 10 CFR 51 10 CFR 70 10 CFR 75 10 CFR 110 10 CFR 140 10 CFR 150 10 CFR 170	9-16-91	12-2-91		Uranium enrichment regulations	Published for comment in 56:179 (46739)
10 CFR 4	3-8-89	5-8-89		Enforcement of nondiscrimination on the basis of handicap in federally assisted programs: notice of proposed rulemaking	Published for comment in 54:44 (9966); corrections in 54:51 (11224)
10 CFR 9			7-15-91; 7-15-91	Duplication fees	Final rule in 56:135 (32070)
10 CFR 11	9-30-91	10-30-91		DOE-L and DOE-Q reinvestigation programs for NRC-R access authorization renewal requirements	Published for comment in 56:189 (49435)
10 CFR 13	9-25-90	11-24-90	9-18-91; 10-18-91	Program Fraud Civil Remedies Act; implementation	Published for comment in 55:186 (39158); corrections in 55:194 (40997); final rule in 56:181 (47132)
10 CFR 16	9-26-90	10-26-90		Salary offset procedure for collecting debts owed by federal employees to the federal government	Published for comment in 55:187 (39285)
10 CFR 19 10 CFR 20 10 CFR 21 10 CFR 30 10 CFR 36 10 CFR 40 10 CFR 51 10 CFR 70 10 CFR 170	12-4-90	3-4-91		Licenses and radiation safety requirements for large irradiators	Published for comment in 55:233 (50008)
10 CFR 20 10 CFR 30 10 CFR 31 10 CFR 34 10 CFR 39 10 CFR 40 10 CFR 70	5-14-90	7-30-90	8-16-91; 10-15-91	Notification of incidents	Published for comment in 55:93 (19890); final rule in 56:159 (40757)

Proposed Rule Changes as of Sept. 30, 1991 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 20 10 CFR 21 10 CFR 73			8-21-91; 9-2-91	Changes in commercial telephone numbers for Region V	Final rule in 56:162 (41448)
10 CFR 20			7-15-91; 7-15-91	Standards for protection against radiation: monitoring reports	Final rule in 56:135 (32071)
10 CFR 21 10 CFR 50			7-31-91; 10-29-91	Criteria and procedures for the reporting of defects and conditions of construction permits	Final rule in 56:147 (36081)
10 CFR 26	8-31-90	10-30-90	8-26-91; 9-25-91	Fitness-for-Duty Programs: nuclear power plant personnel	Published for comment in 55:170 (35648); final rule in 56:165 (41922)
10 CFR 30 10 CFR 40 10 CFR 50 10 CFR 60 10 CFR 61 10 CFR 70 10 CFR 72 10 CFR 110 10 CFR 150	4-3-90	6-18-90		Willful misconduct by unlicensed persons	Published for comment in 55:64 (12374); corrections in 55:70 (13542)
10 CFR 35	1-16-90	4-12-90		Basic quality assurance program, records and reports of misadministration or events relating to the medical use of byproduct material	Published for comment in 55:10 (1439); corrections in 55:25 (4049)
10 CFR 50	3-6-89	7-5-89		Acceptance of products purchased for use in nuclear power plant structures, systems, and components	Published for comment in 54:42 (9229)
10 CFR 50	10-13-89	12-1-89		Nuclear power plant license renewal	Published for comment in 54:197 (41980)
10 CFR 50	10-9-90	12-24-90	8-13-91; 9-12-91	Emergency response data system	Published for comment in 55:195 (41095); final rule in 56:156 (40178)
10 CFR 50	1-31-91	4-16-91		Codes and standards for nuclear power plants	Published for comment in 56:21 (3796)
10 CFR 50			7-10-91; 7-10-96	Monitoring the effectiveness of maintenance at nuclear power plants	Final rule in 56:132 (31306)
10 CFR 50	8-21-91	11-4-91		Decommissioning funding for prematurely shutdown power reactors	Published for comment in 56:162 (41493)
10 CFR 50	8-26-91	10-25-91		Cooperation with states at commercial power plants and other nuclear production or utilization facilities; policy statement	Published for comment in 56:165 (41968)

(Table continues on the next page.)

Proposed Rule Changes as of Sept. 30, 1991 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 51	9-17-91	12-16-91		Environmental review for renewal of operating licenses	Published for comment in 56:180 (47016)
10 CFR 51	7-23-90	10-22-90		License renewal for nuclear power plants; scope of environmental effects	Advanced notice of proposed rulemaking published for comment in 55:141 (29964)
10 CFR 52 10 CFR 71 10 CFR 170 10 CFR 171			7-10-91; 8-9-91	Revision of fee schedules; 100% fee recovery	Final rule in 56:132 (31472); corrections in 56:154 (37828)
10 CFR 55	4-17-90	7-2-90		Operator's licenses	Published for comment in 55:74 (14288)
10 CFR 70 10 CFR 72 10 CFR 73 10 CFR 75	8-15-89	9-29-89		Minor amendments to the physical protection requirements	Published for comment in 54:156 (33570)
10 CFR 71	6-8-88	10-6-88; 12-6-88; 3-6-89; ~6-15-89;[c] 2-9-90		Transportation regulations; compatibility with the International Atomic Energy Agency (IAEA)	Published for comment in 53:110 (21550); corrections published in 53:120 (23484); comment period extended in 53:190 (38297); 2nd extension of comment period in 53:245 (51281); 3rd extension of comment period in 54:63 (13528); comment period end published in 54:237 (51033)
10 CFR 110	2-7-90	3-9-90		Import and export of radioactive wastes	Advance notice of proposed rulemaking for comment in 55:26 (4181); corrections in 55:57 (10786)
10 CFR 110			8-13-91; 8-13-91	Imports from South Africa	Final rule in 56:156 (38335)
48 CFR 20	10-2-89	12-1-89		Acquisition regulation (NRCAR)	Published for comment in 54:189 (40420)

[a]NRC petitions for rule making are not included here, but quarterly listings of such petitions can be obtained by writing to Division of Rules and Records, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Quarterly listings of the status of proposed rules are also available from the same address.

[b]Proposed rules for which the comment period expired more than 2 years prior to the start of the period currently covered without any subsequent action are dropped from this table. Effective rules are removed from this listing in the issue after their effective date is announced.

[c]The expiration date is given as "60 days after the date when the DOT proposed rule is published in the *Federal Register*."

The Authors

Technical Note: A New Approach to Fission Reactor Safety

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Analysis and Modeling of Fission Product Release from Various Uranium–Aluminum Plate-Type Reactor Fuels

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Applications of a Surveillance and Diagnostics Methodology Using Neutron Noise From a Pressurized-Water Reactor

Richard Wood is a research engineer in the Reactor Systems Section of the Instrumentation and Controls Division at the Oak Ridge National Laboratory (ORNL). He received the B.S. and Ph.D. degrees in nuclear engineering from The University of Tennessee, Knoxville. Since joining ORNL he has performed and directed research and development involving data acquisition and analysis, process system diagnosis, thermal–hydraulic modeling, and advanced control system design and analysis. He is currently participating in the evaluation of innovative instrumentation and control systems proposed for use in advanced reactor designs. Current address: ORNL, P.O. Box 2008, Oak Ridge, TN 37831-6010.

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Westinghouse Advanced Passive 600 Plant

Brian A. McIntyre is Manager of the Advanced Plant Safety and Licensing group in the Nuclear and Advanced Technology Division of Westinghouse Electric Corporation. In this position he is responsible for directing the safety and licensing activities for all Westinghouse commercial advanced plant designs, including the AP600. His previous positions include Manager of Safeguards Analysis, with the responsibility for performing coolant accident analyses for Westinghouse reactor designs. Upon joining Westinghouse in 1972, McIntyre worked on designing large-scale heat transfer tests for new emergency core cooling system designs. He received the B.S.ME and M.S.ME degrees from Purdue University and is a member of the American Society of Mechanical Engineers; he has served on various NUMARC and Institute of Nuclear Power Operations committees. Current address: Westinghouse Electric Corporation, Advanced Plant Safety and Licensing, Box 355, Pittsburgh, PA 15230-0355.

Rita K. Beck is a Senior Engineer in the Advanced Plant Safety and Licensing group in the Nuclear and Advanced Technology Division of Westinghouse Electric Corporation. In this position she is responsible for coordinating the licensing aspects of the AP600 design certification. Her previous positions include plant and systems evaluations and licensing, coordinating design basis documentation programs, and nuclear operations instruction. Beck has a Senior Reactor Operator Instructor certification from the Nuclear Regulatory Commission and received the B.S. degree in mining engineering from Montana College of Mineral Science and Technology. Current address: Westinghouse Electric Corporation, Advanced Plant Safety and Licensing, Box 355, Pittsburgh, PA 15230-0355.

System 80+™ PWR Safety Design

C. W. Bagnal is Assistant Project Manager, Nuclear Systems Development, ABB Combustion Engineering (CE) Nuclear Power. Bagnal earned the Bachelor of nuclear engineering degree and a Certificate in Materials Engineering from Georgia Institute of Technology. He has 15 years experience developing advanced nuclear power reactors and refueling schemes for 22 utilities. Bagnal developed practical fuel management techniques to reduce reactor vessel embrittlement and resolve the pressurized thermal shock issue for CE reactors, managed extended burnup reload design and licensing for two plants, conducted reactor analysis workshops for nine utilities, and contributed to efforts that docketed the first advanced light-water reactor in the United States under new Nuclear Regulatory Commission regulations for future nuclear plants (10 CFR 52) in 1991. Current address: ABB-CE Nuclear Power, 1000 Prospect Hill Rd., P. O. Box 500, Windsor, CT 06095-0500.

R. A. Matzie is Vice President, Nuclear Systems Development, ABB Combustion Engineering (CE) Nuclear Power. Matzie is responsible for the development, licensing, and marketing of advanced reactors, including System 80+ and the commercial MHTGR. He is also a Captain in the U.S. Navy Reserve. Matzie is a graduate of the U.S. Naval Academy and earned the Ph.D. degree in nuclear engineering from Stanford University. He served in the Navy nuclear submarine program for 5 years and has been with ABB-CE since 1975, holding various technical and management positions, including Manager of Analog Plants, Manager of Reactor Engineering, and Director of Advanced Nuclear Systems. His main areas of technical interest are the development of advanced nuclear systems and fuel cycles. Matzie has published more than 60 technical papers in industry journals. Current address: ABB-CE Nuclear Power, 1000 Prospect Hill Rd., P. O. Box 500, Windsor, CT 06095-0500.

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The MATS Experiments—Mesoscale Atmospheric Transport Studies at the Savannah River Site

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S. Berman holds the position of Professor and Program Coordinator for Meteorology in the Earth Sciences Department at the SUNY-College at Oneonta in Oneonta, New York. He received his Ph.D. degree in meteorology from the University of Wisconsin in 1972, studying coherence properties of surface-layer winds under the supervision of Dr. Heinz, H. Lettau, and Dr. Charles R. Stearns. Berman has maintained an active research interest in atmospheric turbulence and diffusion problems through his summer affiliations with national laboratories at Argonne and Oak Ridge and more recently with the Environmental Transport Section at Department of Energy's Savannah River Laboratory at the Savannah River Site near Aiken, S.C. Current address: Earth Sciences Department, SUNY-College at Oneonta, Oneonta, NY 13820-4015.

R. Kurzeja is a Principal Scientist at the Westinghouse Savannah River Company. He holds a B.A. degree from the University of Chicago in geophysics and M.S. and Ph.D. degrees from Florida State University in meteorology. He was a Research Professor at George Washington University's Department of Engineering and Applied Sciences prior to his present position. He has worked in atmospheric chemistry, atmospheric dynamics, and atmospheric dispersion. Current address: Building 773-A, Rm. A-1118, Savannah River Laboratory, Aiken, SC 29802.

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Book Review: *Environmental Radioactivity in the European Community 1984–1985–1986*

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Aging Assessment of BWR Control Rod Drive Systems

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Indexes to *Nuclear Safety*, Volume 32

Cumulative indexes of issues of *Nuclear Safety* through Volume 22 were published as separate documents and are available from the National Technical Information Service. Starting with Volume 23, author and KWIC indexes are published in the first issue of the following volume. Thus this issue contains these indexes for *Nuclear Safety*, Vol. 32. Both indexes use a seven-digit number to indicate the location of the indexed material. The seven-digit number is divided into four parts (00-0-0-000), which stand, respectively, for volume-number-section-page. The authors are indexed alphabetically. In the KWIC (Key Word in Context) index, the article titles are permuted around the various significant words contained therein. For example, the title "Design Basis for Nuclear Power Plant Protection Systems" is indexed under the words Design, Nuclear, Power, Plant, and Protection. The index words are arranged alphabetically in a column in the center of the page, with the titles permuted around them. A slash (/) indicates the end of a title. The two indexes follow.

KWIC Index

Author List

ERRATA

In *Nuclear Safety,* Vol. 32, No. 4, several errors occurred in the article "Indoor Radon: A Natural Risk" by N. H. Harley and J. H. Harley (pages 537 to 543):

1. John H. Harley was erroneously shown as "Consultant for the Environmental Measurements Laboratory." The identification should have read simply "Consultant."
2. Naomi H. Harley's current address should have read: Department of Environmental Medicine, New York University School of Medicine, 550 First Ave., New York, NY 10016, rather than the address shown under *The Authors* (page 620).
3. In two places several lines of text were inadvertently omitted from the article. The omitted text is shown below in bold-face type:

3.1 Page 537, lower right-hand corner:

MINER EXPERIENCE

Experience with several groups of underground miners has shown a significant excess of lung cancer among those exposed to radon, even when smoking habits are taken into consideration. The risk comes **from the short-lived decay products deposited on the bronchial epithelium of the respiratory tract during inhalation and exhalation. The two isotopes of polonium** in this chain are alpha emitters and can deliver a considerable radiation dose to cells lining bronchial airways. These areas are the major site of lung cancer in the miners as well as in those exposed to other carcinogens through smoking.

3.2 Page 538, left-hand corner:

RISK ESTIMATION FOR MINERS

The only data we have for risk assessment is that from lung cancer in underground miners. There are no definitive studies from population exposures, and there is no strong evidence that radon exposure causes other cancers or other diseases. There is a pattern in the miner mortality that risk is generally proportional to cumulative exposure, that lung cancer does not appear before age 40 (the normal age for appearance of lung cancer), and that there is a minimum latent period of about 5 yr before lung cancer appears in an exposed individual.

The cancer data for the four major groups of miners **are shown in Table 1. Collateral data from other miners (Howe et al.,[13] Morrison et al.,[14] Samet et al.[15]) reinforce these data and show both a dose-response relationship and a comparable radon risk for mines with widely different atmospheres. Most of these miners are no longer working in the mines but are being followed carefully because the present-day miners are exposed to much lower concentrations and will probably not contribute a significant number of new lung cancers to the studies.**

While we have fairly up-to-date follow-up information on the miner populations, we do not have lifetime mortality risks because a large fraction of the largest groups are still alive. Extrapolating the available data to lifetime mortality has several major problems, including the following.

4. Page 539, the 6th line in the section headed "Population Exposures" should read: "The global release is about 100 EBq/yr (3 GCi/yr),. . ."
5. Page 540, the penultimate paragraph should begin: "In the case control study of New Jersey women. . ." rather than "In the case of a control study. . ."
6. Page 541, the last paragraph in the section headed "Guidelines" should begin "While the limits for indoor radon. . .", rather than "Because the limits for indoor radon. . ."
7. Page 542, the last sentence of the first paragraph of the section headed "Conclusions" should read: "This is not necessarily due to failure of the linearity concept, but due to other factors not presently understood." rather than "This is not necessarily due to failure of the linearity concept, but because other factors are not presently understood."

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