NRC PDR

INDEX OF BRIEFING PAPERS

BP18 Environmental Effects of Nuclear Power Plants
[NRR]

BP19 Uranium Mill Tailings
[NMSS]

BP20 State Compliance with the 1993 and 1996 Milestones of the Low Level Radioactive Waste Policy Amendment Act of 1985
[NMSS]

BP21 Incident Investigation Teams
[AEOD]

BP22 Rosemont Transmitters
[NRR]

BP23 Harassment and Intimidation in the Nuclear Industry
[OE]

BP24 Decommissioning of Nuclear Power Plants
[NRR]

BP25 BWR Water Level Instrument Errors Due to Noncondensible Gas
[NRR]

BP26 Fire Barrier Issues
[NRR]

BP27 Steam Generator Tube Plugging Criteria
[NRR]

BP28 Medical Issues
[NMSS]

BP29 Integrity of Reactor Vessel Internals
[NRR]

LICENSE RENEWAL

Background:

Nuclear power provides approximately 20 percent of the electric power produced in the United States. The term of the initial operating license for a nuclear power plant is limited to 40 years. The first plant's 40-year operating license will expire in the year 2000 and approximately 20 percent of the rest will expire by the end of the year 2010. The Department of Energy estimates that by the year 2000, significant new electric generating capacity will be needed. The timely renewal of these operating licenses for an additional 20 years, where appropriate to do so, represents an important contribution to ensuring an adequate energy supply for the nation during the first half of the 21st century.

License Renewal:

The Atomic Energy Act limits commercial power reactor licenses to 40 years, but also permits the renewal of such licenses. The technical steps, the procedural steps, and the criteria to determine if regulatory requirements are met for license renewal are established in 10 CFR Part 54 and 10 CFR Part 51. License renewal is based on two key principles. The first principle is that the regulatory process is adequate to ensure that the licensing basis of all currently operating plants provides an acceptable level of safety, with the exception of age-related degradation unique to license renewal and possibly some few other issues during extended operation. The second principle is that each plant's current licensing basis is required to be maintained during the renewal term including management of aging of systems, structures, and components needed to ensure safe operation and accident mitigation. In other words, the foundation of license renewal rests on the determination that currently operating plants were initially shown to have adequate levels of safety and this level has been enhanced through evolution of the licensing bases. Additionally, NRC regulatory activities have provided ongoing assurance that the licensing bases continue to provide an acceptable level of safety.

Specifically, a nuclear power plant may apply to the Commission to renew its license for a period of 20 years or less. This application would be subject to public hearings - a formal, adjudicatory process. The license renewal review will be based on issues primarily related to aging management and on how the renewal application addresses the effects of degradation that would occur in the period of extended operation. The Commission expects the initial license renewal review process to take approximately 5 years based on a detailed technical review and hearing process. The Commission estimates that an applicant would need approximately 3 to 5 years to prepare its application. An applicant may apply as early as 20 years before the expiration of its current license.

Regulations:

The decision whether to seek license renewal rests with the licensees. They must make business decisions as to whether they are likely to satisfy NRC requirements and evaluate the costs that will be incurred to do so. NRC's task is to establish a reasonable process and safety standards so that they can make timely decisions whether to seek license renewal.

The Commission's regulations governing license renewal are contained in 10 CFR Part 54 and require the applicant to describe and justify how they will identify and screen all the systems, structures, and components (SSCs) important to license renewal. These SSCs include all safety related SSCs, all SSCs whose failure could directly affect safety related functions, and all SSCs subject to operability requirements contained in the facility's technical specifications limiting conditions for operation. Further, the regulations include SSCs relied on to demonstrate compliance with the Commission's regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout. The regulations recognize that in the screening process, applicants may identify areas in which new or modified programs will be developed to assess and manage the effects of age-related degradation.

A parallel rulemaking effort concerning environmental issues related to license renewal is presently in progress. The public comment period for the draft Generic Environmental Impact Statement (GEIS) and the proposed rule revision to 10 CFR Part 51 ended in March 1992. This rulemaking is based on the belief that certain environmental issues should be treated generically, rather than in each plant specific licensing review. The public comments on the draft GEIS and proposed rule have raised concerns related to both procedural aspects of the rule and NRC policy for treatment of environmental issues. The Commission has reached agreement with the Council on Environmental Quality and the Environmental Protection Agency concerning the procedural concerns and expects to resolve the remaining policy issues in the near future. As directed by the Commission in a staff requirements memorandum dated April 22, 1993, the staff will conduct discussions (in the form of workshops) with the commenters to resolve the policy issues. These workshops will be held in locations convenient to the commenters but are not firmly scheduled. The Commission's resolution of the procedural aspects of the rule and a discussion of the policy issues were made public in SECY-93-032, "10 CFR Part 51 Rulemaking on Environmental Review for Renewal of Nuclear Power Plant Operating Licenses," dated February 9, 1993.

Industry Activities:

The NRC staff has reviewed a series of industry technical reports on evaluation of age-related degradation effects on a variety of structures and components important to license renewal. The staff plans to incorporate appropriate technical information from the industry technical reports into the standard review plan for license renewal.

In 1989, the Department of Energy and the Electric Power Research Institute selected Yankee Rowe and Monticello to be lead plants to demonstrate the license renewal process for the industry. The Northern States Power Company (operator of Monticello) had been preparing a renewal application until late 1992 when the effort was placed on indefinite hold. Economic and regulatory reasons were cited as determining factors. Also in 1992, concerns regarding the integrity of Yankee Rowe's reactor pressure vessel led to a decision to suspend operations rather than seek license renewal.

Current Status:

Since publication of the final license renewal rule, a number of significant policy issues have been identified including the question of whether the maintenance and license renewal rules can be integrated further, the appropriate scope of the license renewal rule, and the appropriate interpretation of age-related degradation unique to license renewal. The staff, in SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses For Nuclear Power Plants'," of March 1, 1993, and SECY-93-113, "Additional Implementation Information for 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants'," of April 30, 1993, proposed approaches for implementing the license renewal rule to resolve industry concerns about the integrated plant assessment and discussed the resolution of a variety of other issues the industry identified since the final license renewal rule (10 CFR Part 54) was published in December, 1991. After considering the staff proposals, the Commission directed the staff to convene a public workshop to examine the extent to which greater reliance can be placed on the maintenance rule and other existing licensee activities and programs for purposes of license renewal. The NRC staff held the workshop on September 30, 1993, the results of which were summarized in a Commission paper, SECY-93-331, "License Renewal Workshop Results and Staff Proposals for Revision to 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants'," dated December 7, 1993. In SECY-93-331, the staff presented its conclusions and proposals regarding an approach to license renewal that allows greater credit for existing licensee programs and maintenance rule requirements in the license renewal process, and recommended specific changes to the license renewal rule to reflect these proposals. The NRC staff's proposal in SECY-93-331 focuses on managing the effects of age-related degradation rather than identifying and managing age-related degradation mechanisms.

Currently, several individual industry efforts are underway that have progressed to the stage of interaction with the NRC. The Babcock & Wilcox (B&W) Owners Group, representing seven operating B&W units, has formulated a generic license renewal program and the Baltimore Gas and Electric Company (BG&E) has developed a specific approach for its Calvert Cliffs plant. Both B&W Owners Group and BG&E submitted methodolgies for screening system, structure, and components important to license renewal, and the staff's review of these methodologies is nearing completion. The Westinghouse Owners Group recently met with the staff to describe a 5 year program for life cycle management/license renewal. The Nuclear Management and Resources Council (NUMARC) is the leading industry group interacting with the NRC on license renewal policy issues. In two letters to the Commission, dated October 12 and November 18, 1993, NUMARC presented a proposal to resolve industry concerns.

Contact:

Scott Newberry, Director, License Renewal and Environmental Review Project Directorate, Associate Directorate for Advanced Reactors and License Renewal, Office of Nuclear Reactor Regulation, U.S. NRC, Washington, D.C. 20555. (301) 504-1183.

HIGHLIGHTS OF LICENSE RENEWAL

- Nuclear power provides approximately 20 percent of the electric power produced in the United States.

- License renewal could represent an important contribution to the nation's future energy supply.

- Atomic Energy Act limits initial licenses to 40 years but allows for renewal of the license. Code of Federal Regulations 10 CFR Part 54 allows for license renewal up to 20 years and provides procedures and requirements for license renewal applications.

- Application review focused on the effects of age-related degradation of structures, systems, and components important to license renewal.

- Generic Environmental Impact Statement and amendment to 10 CFR Part 51 address environmental issues associated with license renewal of nuclear power plants.

- Applicants are encouraged to submit their applications at least 5 years prior to expiration of current license.

- Yankee Rowe's decision to decommission announced February 1992. Monticello announced an indefinite delay in making a decision on whether or not to seek a renewed license. (November 1992)

- Babcock and Wilcox Owners Group is pursuing generic resolution of license renewal issues affecting the B&W plant design.

- Baltimore Gas and Electric Company is currently working with staff to resolve license renewal issues for their Calvert Cliffs Nuclear Power Plant.

- Commission papers, SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licensees For Nuclear Power Plants'," of March 1, 1993, and SECY-93-113, "Additional Implementation Information for 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants'," of April 30, 1993, describe the staff's proposed approach for implementing the integrated plant assessment without a rule change and discusses significant license renewal issues.

- In SECY-93-049 the staff proposed to incorporate the areas of technical agreement in the industry technical reports into the standard review plan for license renewal instead of in SERs. The Commission directed this be done in the SRM dated June 28, 1993.

- As directed in the Commission's SRM dated June 28, 1993, the NRC staff held a public workshop on September 30, 1993, to receive industry and public input on how best to resolve license renewal issues.

- In SECY-93-331, the NRC staff presented its conclusions and proposals regarding an approach to license renewal that allows greater credit for existing licensee programs and maintenance rule provisions in the license renewal process, and recommended specific changes to the license renewal rule to reflect these proposals. The staff's proposal emphasizes managing the <u>effects</u> of aging rather managing aging <u>mechanisms</u>.

DESIGN CERTIFICATION PROCESS

Background:

The Commission has long sought nuclear power plant standardization and the enhanced safety and licensing reform which standardization could make possible. The Commission has also sought to improve the licensing environment for advanced nuclear power reactors by minimizing the uncertainty in the regulatory process. To do this, the Commission has promulgated 10 CFR Part 52 (54 FR 15372; April 18, 1989) which sets out a sensible and stable procedural framework for standardization. This framework involves early design consideration and certification, through rulemaking, of future designs. The design certification process is the key procedural device in the Commission's regulations for bringing about the sought-after goal of enhanced safety and early resolution of licensing issues. The goals and objectives of Part 52 have been reaffirmed in the Energy Policy Act of 1992.

Design Certification:

The Associate Directorate for Advanced Reactors and License Renewal was established, in part, to provide a focal point for early staff interactions with the sponsors of advanced reactor plant designs. The review process for advanced reactor plant designs leading to design certification is specified in Subpart B of 10 CFR Part 52. These reviews are expected to result in the certification of individual designs following the completion of a rulemaking process. Design certifications are issued for a duration of fifteen years.

Regulations:

The Commission's regulations governing the design certification process are contained in 10 CFR Part 52 and require an applicant to provide the technical information necessary to demonstrate compliance with the standards set out for construction permits and operating licenses in 10 CFR Parts 20, 50, 73, and 100, as those standards are technically relevant to the design of the proposed facility. Subpart B of 10 CFR Part 52 sets out the requirements and procedures applicable to Commission issuance of rules granting standard design certification for nuclear power facilities separate from the filing of an application for a construction permit or combined license for such a facility. Appendix O of 10 CFR Part 52 sets out procedures for the filing, staff review, and referral to the Advisory Committee on Reactor Safeguards of standard designs for an advanced nuclear power reactor.

A key provision of 10 CFR Part 52 relates to the involvement of the public in the design certification process. Public participation is more effective under Part 52 because the bulk of the issues are considered in the design certification part of the process.

Under 10 CFR Part 52, applicants must provide information related to the Three Mile Island Requirements set forth in 10 CFR 50.34, the postulated site parameters, the resolution of unresolved and generic safety issues, and design-specific probabilistic risk assessment. Also, applicants must provide inspection, tests, analyses, and acceptance criteria (ITAAC) and interface requirements.

Current Status:

The status of advanced light water reactor reviews are based on the schedules presented in SECY-93-097, "Integrated Review Schedules for the Evolutionary and Advanced Light-Water Reactor Projects." Specifically:

GE Nuclear Energy (GE) Advanced Boiling Water Reactor (ABWR): The final design approval is expected to be issued in 1994 following the review of the staff's safety evaluation report by the Commission and ACRS.

ABB-Combustion Engineering System 80+: The staff's safety evaluation report is expected to be issued for Commission and ACRS review by the end of February 1994.

Westinghouse AP600: The new features of the design are undergoing an intensive testing and verification process by both Westinghouse and the NRC staff. The staff is conducting the detailed design review and expects to issue a Draft Safety Evaluation Report (DSER) after the majority of the test program is complete.

GE Nuclear Energy Simplified Boiling Water Reactor (SBWR): The staff completed an acceptance review of the application submitted by GE and determined that it was acceptable in May 1993. The staff is conducting the detailed design review and expects to issue a DSER after the majority of a verification test program is complete.

Contact:

R. W. Borchardt, Director, Standardization Project Directorate, Associate Directorate for Advanced Reactors and License Renewal, Office of Nuclear Reactor Regulation, U.S. NRC, Washington, D.C. 20555. (301) 504-1118.

HIGHLIGHTS OF DESIGN CERTIFICATION PROCESS

- The Commission has long sought nuclear power plant standardization.

- Standardization will enhance safety.

- Licensing environment improved by minimizing uncertainty.

- Design certification is the key procedural device for bringing about early resolution of safety issues.

- Design certification review process specified in Subpart B of 10 CFR Part 52.

- Design certification reviews assure compliance with standards set out for construction and operating licenses in 10 CFR Parts 20, 50, 73, and 100.

- 10 CFR Part 52 makes public participation more effective by moving the bulk of the issues up front to the design certification stage.

- Applicants must provide information related to the Three Mile Island requirements, the postulated site parameters, the resolution of unresolved generic safety issues, design-specific probabilistic risk assessment, ITAAC, and the interface requirements.

- The goals and objectives of Part 52 were reaffirmed in the Energy Policy Act of 1992.

- Commission briefing on the status of advanced reactors reviews - June 2, 1993 (SECY-93-097).

- GE Nuclear Energy Advanced Boiling Water Reactor final design approval is expected to be issued in 1994 following the review of the staff's safety evaluation report by the Commission and ACRS.

- ABB-Combustion Engineering System 80+ final safety evaluation report is expected to be issued for Commission and ACRS review by the end of February 1994.

- Westinghouse AP600 application was accepted December 1992, and the design certification process, including an intensive testing program, is on-going.

- GE Nuclear Energy simplified boiling water reactor application was accepted May 1993, and the design certification process is on-going.

BELOW REGULATORY CONCERN

Background:

On July 3, 1990, the Commission published a Below Regulatory Concern (BRC) Policy Statement in the Federal Register. The BRC Policy was intended to guide a broad range of Commission actions, including exemptions from Commission regulations, as well as the development of generic health and safety standards such as those involved in the rulemaking on radiological criteria for decommissioning. Subsequent to the publication of the BRC Policy, the Commission placed an indefinite moratorium on the implementation of the BRC Policy because of the broad public concern expressed over the new Policy. Section 2901 of the recently enacted National Energy Policy Act of 1992 (H.R. 1776) revoked the Commission's July, 1990, BRC Policy Statement. Section 2901 also revoked the Commission's policy statement of August 29, 1986 that established criteria to guide Commission exemption decisions on specific low-level radioactive waste streams. This latter policy was developed in order to comply with Section 10 of the Low-level Radioactive Waste Policy Amendments Act of 1985. The Commission issued a formal withdrawal of these two policy statements in the Federal Register on August 24, 1993 (58 FR 44610).

BRC Policy Statement:

The BRC Policy was intended to provide a framework for making decisions on whether to grant specific exemptions in categories such as: (1) the cleanup or release of sites containing residual radioactivity; (2) the distribution of consumer products containing small amounts of radioactivity; (3) the disposal of certain wastes containing very low levels of radioactivity; or (4) the recycling or reuse of radioactive materials that have very low levels of radioactivity. Materials in the above categories with sufficiently low levels of radiation would be exempt from regulatory controls.

However, the issuance of the BRC Policy resulted in extensive comment and public concern. The public reaction resulted in the introduction of legislation on the national level, as well as by a number of State and local governments, that would prevent the BRC Policy from taking effect.

Current Status:

After the Commission placed the indefinite moratorium on the implementation of the BRC Policy, it decided to initiate rulemaking to address the critical need for generic site cleanup and decommissioning standards for NRC-licensed facilities. The

Commission determined that it should proceed with a fresh approach to the development of these standards that is independent of the now defunct BRC Policy. The Commission intends to enhance the participation of affected interests in the rulemaking by soliciting commentary from these interests on the rulemaking issues before the staff develops the draft proposed rule. The Commission has just completed a series of workshops to solicit commentary from affected interests on the fundamental approaches and issues that must be addressed in establishing the radiological criteria for decommissioning. The workshops were held in various locations throughout the United States beginning in January, 1993 and were open to the public. The NRC staff will use the workshop comments in the development of a draft proposed rule on the site cleanup criteria for decommissioning. A draft of the NRC staff version of the criteria will be available for comment by workshop participants and others in January, 1994. It is anticipated that the Commission will issue the proposed rule for public comment in June, 1994.

Contacts:

Francis X. Cameron, OGC (301) 504-1642
Donald A. Cool, RES (301) 492-3785
Cynthia G. Jones, NMSS (301)504-2629
Michael F. Weber, NMSS (301)504-1298

HIGHLIGHTS

- The BRC Policy Statement was an attempt to establish a framework for making decisions on whether to grant specific exemptions from Commission regulations.

- The categories considered for possible specific exemptions were:

  (1) cleanup or release of sites for residual radioactivity;

  (2) distribution of consumer products containing small amounts of radioactivity;

  (3) disposal of slight, contaminated wastes; or

  (4) recycling or reuse of radioactive materials that have very low levels of radioactivity.

- The BRC Policy Statement was issued on July 3, 1990 in the Federal Register.

- In response to public concern, the Commission declared a moratorium on the implementation of the BRC policy on June 28, 1991.

- With this moratorium, NRC initiated a phased consensus-building process on BRC issues.

- On November 12, 1991, potential representatives of environmental interests informed NRC that it would be unable to represent the environmental community in the BRC consensus process. Due to the inability to bring all affected interests to the table, the Commission decided to abandon its effort to build consensus on the BRC Policy. However, the Commission continued its indefinite moratorium on the implementation of the BRC Policy.

- The Commission issued a Federal Register Notice on August 24, 1993 (58 FR 44610) that formally withdrew the BRC Policy of 1986 and the BRC Policy of 1990 in accordance with the National Energy Policy Act of 1992.

- As an alternate to the consensus process, the Commission initiated an enhanced participatory rulemaking that would provide early access to affected interests on the development of radiological criteria for decommissioning.

- The NRC staff is now developing a proposed rule on the site cleanup criteria for decommissioning utilizing the results of several public workshops held throughout the United States in 1993.

RESIDENT INSPECTOR PROGRAM

Background:

The resident inspection program is an important element of the NRC mission to protect public health and safety. The responsibility for safe operation of a nuclear power plant lies with the licensee. The NRC inspection program is designed to make selective examinations to ensure that this responsibility is being met. The NRC inspection program is oriented toward audits; thus, it does not examine every activity or item, but attempts to verify through carefully selected samples, that the activities are being properly conducted or operated to enhance or ensure safety.

Resident Inspectors:

In 1977, the NRC initiated a program to station resident inspectors at each nuclear power plant under construction and in operation. Since that time, the program has expanded to the point where normally at least two resident inspectors are assigned to each site with a nuclear power plant.

The onsite resident inspectors live in the area of the nuclear power plant. They maintain offices at the plant and are normally available during regular business hours. In addition, resident inspectors spend a portion of their time at the plant during weekends and evenings. By assigning resident inspectors to reactor sites, the NRC was able to significantly increase the amount of time inspectors spend at the plant. This increased time provides a greater opportunity to observe and measure licensee activities, verify licensee compliance with NRC requirements, and respond to operational events at the plant. Since a resident inspector is assigned to a single site, the resident inspector acquires more detailed knowledge of that plant and is able to provide more efficient inspections.

The resident inspector provides a continual inspection and regulatory presence, as well as a direct contact between NRC management and the licensee. The resident inspector is also the key individual in the regional offices' determination of what additional inspection activities need to be accomplished at a specific plant. The inspection activities of the resident inspector are supplemented by the efforts of engineers and specialists from the regional office staff who perform inspections in a wide variety of engineering and scientific disciplines ranging from civil and structural engineering to health physics and core physics.

Regulations:

The U.S. Atomic Energy Commission (AEC) was created by an act of Congress in 1946. In the Atomic Energy Act of 1954, the AEC was vested with developmental and regulatory functions related to peaceful uses of atomic energy. In 1974, another law was passed (the Energy Reorganization Act) which abolished the Atomic Energy Commission and created two new organizations, the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC).

The NRC has broad authority under the Atomic Energy Act of 1954, as amended (Act). This authority is reflected in Section 161o of the Act which provides authority for inspections. This authority has been implemented through the promulgation of regulations, specifically 10 CFR 50.70, which requires nuclear power plant licensees to permit inspections deemed necessary by the NRC.

Current Status:

In Fiscal Year 1994, there were 107 operating nuclear power plants, 2 plants that have been shut down indefinitely, and 7 plants with construction permits. For these plants, NRC budgeted 181 resident inspectors and 188 regional office inspectors.

Contact:

Mark W. Peranich, Chief, Operating Reactor Inspection Section
Inspection and Licensing Policy Branch
US NRC, Washington, DC 20555. (301) 504-3078

HIGHLIGHTS OF RESIDENT INSPECTOR PROGRAM

- Licensee is primarily responsible for all safety, safeguards, and environmental measures necessary to protect the public health and the environment.

- The NRC role is to determine how well the licensee is performing and to ensure that the licensee corrects poor performance whenever identified.

- Section 161o of the Atomic Energy Act of 1954, as amended, provides authority for inspections. This authority has been implemented through 10 CFR 50.70, which requires nuclear power plant licensees to permit inspections deemed necessary by the NRC.

- In 1977, the NRC initiated the resident inspection program

- The resident inspection program has provided:

  - Increased NRC knowledge of the conditions at licensed nuclear power plants and a better technical base for regulatory actions.

  - Lessened reliance on the accuracy and completeness of licensee records by improving the inspector's ability to independently verify licensee performance.

  - Additional assurance that licensee management control systems are effective and licensee performance is acceptable.

- As of 1994, there were 107 operating nuclear power plants, 2 plants that have been shut down indefinitely, and 7 plants with construction permits. For these plants, NRC budgeted 181 resident inspectors and 188 regional office inspectors.

ASSISTANCE TO REGULATORY BODIES OF RUSSIA AND UKRAINE

Background:

The NRC has played a leading role during the last five years in cooperative efforts under the Joint Coordinating Committee for Civilian Nuclear Reactor Safety (JCCCNRS) to improve safety in the former Soviet Union, mainly through bilateral information exchanges, joint technical working groups, exchanges of safety inspectors and, recently, in specific coordinated research.

Following the breakup of the USSR in late 1991, relationships with the former republics were maintained as they had previously with the Soviet Union. For the NRC, these relationships devolved to Russia and Ukraine inasmuch as they were the only former republics with operating nuclear power reactors. (Lithuania, which has a reactor plant, is considered as part of Eastern Europe for assistance purposes.)

Current Status:

The seriousness of safety concerns for Soviet-designed reactors led to a major international effort to organize assistance to the newly independent states. In May 1992, then-Secretary of State Baker announced in Lisbon, Portugal, a major program of US assistance for Russia and Ukraine. A significant component of this program was nuclear safety. Most of the nuclear safety projects proposed derived from the previous experience with the JCCCNRS activities.

The four components of the nuclear safety portion of the Lisbon Initiative are:

- Establishment of two regional training centers, one in Russia and the other in Ukraine;

- Immediate operational safety enhancements for certain reactor types;

- Risk reduction measures for other reactor types; and,

- Regulatory assistance in developing consistent and effective safety standards and procedures, as well as training in their use.

Congress approved $25 million in FY 92 funds to initiate work on these proposals, with $3.1 million to support NRC regulatory assistance activities.

In July of 1992, the heads of the US, Russian and Ukrainian regulatory agencies met in the US to reach agreement on the action needed to implement the regulatory assistance component. Twenty projects were identified to assist in the development of legislative, regulatory and liability frameworks for effective, independent regulatory programs.

In September 1992, the NRC signed an Inter-Agency Agreement with the United States Agency for International Development to provide direct assistance to the Russian and Ukrainian regulatory bodies. The United States DOE signed a similar agreement to provide assistance under the Lisbon Nuclear Safety Initiative for the Newly Independent States. Implementation began in October 1992 and will last for about three years.

In March 1993 at the annual JCCCNRS meeting in Kiev, Ukraine, the JCCCNRS organizational structure was formally recognized to include the Lisbon Initiative. This meeting was significant in that it marks the first joint meeting of the US-Russia and US-Ukraine JCCCNRS. The meeting officially established two separate JCCCNRS's, meeting annually on a trilateral basis, the designation of Co-Chairmen from each country and the joint participation in cooperative and other selected activities. In addition all parties agreed to extend the Memorandum of Cooperation in the Field of Civilian Nuclear Reactor Safety for another 5 years.

Currently, activities are in progress to ensure effective and timely implementation of the President's Vancouver Summit Assistance to Russia.

Additional project funding ($5 million) became available with the signing of the FY-93 IAA in June 1993 to support continuation of the NRC activities.

Contact:

Jacob F. Wechselberger, Technical Program Manager for International Activities, Office of the Executive Director for Operations, USNRC, Washington, DC 20555, (301) 504-1730

EMERGENCY PLANNING AND PREPAREDNESS

Background:

Following the accident at Three Mile Island in 1979, the Nuclear Regulatory Commission (NRC) reexamined the role of emergency planning for protection of the public in the vicinity of nuclear power plants. The Commission issued regulations requiring that before a plant could be licensed to operate, the NRC must have "reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." The regulations set forth 16 emergency planning standards and define the responsibilities of licensees and State and local organizations involved in emergency response.

Emergency Planning and Preparedness:

Emergency planning has been adopted as an added conservatism to the NRC's "defense in depth" safety philosophy. Briefly stated, this philosophy: (1) requires high quality in the design, construction and operation of nuclear plants to reduce the likelihood of malfunctions in the first instance; (2) recognizes that equipment can fail and operators can make mistakes, therefore requiring safety systems to reduce the chances that malfunctions will lead to accidents that release fission products from the fuel; and (3) recognizes that, in spite of these precautions, serious fuel damage accidents can happen, therefore requiring containment structures and other safety features to prevent the release of fission products offsite. The added feature of emergency planning to the defense in depth philosophy provides that, even in the unlikely event of an offsite fission product release, there is reasonable assurance that emergency protective actions can be taken to protect the population around nuclear power plants.

Regulations:

For planning purposes the Commission has defined a plume exposure pathway emergency planning zone (EPZ) consisting of an area about 10 miles in radius and an ingestion pathway EPZ about 50 miles in radius around each nuclear power plant. EPZ size and configuration may vary in relation to local emergency response needs and capabilities as affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

The Commission's 16 emergency planning standards are contained in 10 CFR Part 50.47. They cover the following topics:

1. Assignment of Responsibility
2. Onsite Emergency Organization
3. Emergency Response Support and Resources
4. Emergency Classification System
5. Notification Methods and Procedures
6. Emergency Communications
7. Public Education and Information
8. Emergency Facility and Equipment
9. Accident Assessment
10. Protective Response
11. Radiological Exposure Control
12. Medical and Public Health Support
13. Recovery and Reentry Planning and Post-Accident Operations
14. Exercises and Drills
15. Radiological Emergency Response Training
16. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans

Detailed information about emergency planning and preparedness is contained in Appendix E of 10 CFR Part 50 and in NUREG-0654 (FEMA-REP-1), a joint publication of the NRC and the Federal Emergency Management Agency (FEMA) entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

Current Status

In the U.S., commercial nuclear power reactors are currently licensed to operate at approximately 70 sites in about 30 states. For each there are onsite and offsite emergency plans to assure that adequate protective measures are taken to protect the public in the event of a radiological emergency. Federal oversight of emergency planning for licensed nuclear power plants is shared by the NRC and FEMA through a memorandum of understanding. The memorandum is responsive to the President's decision of December 7, 1979, that FEMA will take the lead in offsite planning and response, his request that NRC assist FEMA in carrying out this role, and the NRC's continuing statutory responsibility for the radiological health and safety of the public.

Each licensee at each site exercises its emergency plan with offsite authorities so that State and local government emergency plans for each operating reactor site are exercised biennially, with participation of State and local governments, within the plume exposure EPZ.

CONTACT: Falk Kantor, Emergency Preparedness Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, (301) 504-3924

HIGHLIGHTS OF EMERGENCY PLANNING AND PREPAREDNESS

- Three Mile Island accident focused attention on emergency planning.
- NRC must have reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.
- NRC regulations in 10 CFR 50.47 contain 16 emergency planning standards.
- Emergency planning is part of NRC's "defense in depth" safety philosophy.
- The plume exposure planning zone (EPZ) extends about 10 miles in radius around each licensed nuclear power plant.
- The ingestion pathway EPZ extends about 50 miles in radius.
- Details about emergency planning are contained in Appendix E of 10 CFR Part 50 and in NUREG-0654.
- The NRC and the Federal Emergency Management Agency (FEMA) share federal oversight of emergency planning for licensed nuclear power plants through a memorandum of understanding.
- Nuclear power reactor licensees exercise their emergency plans with those of offsite authorities biennially.

EMERGENCY RESPONSE DATA SYSTEM

Background:

As a result of the accident at Three Mile Island, Unit 2, on March 28, 1979, the Nuclear Regulatory Commission (NRC) and others recognized the need to substantially improve the NRC's ability to acquire accurate and timely data on plant conditions during emergencies. The NRC's role in the event of an emergency is primarily to monitor the licensee to ensure that appropriate recommendations are made regarding offsite protective actions. The NRC provides the licensee with technical analysis and logistic support, supports offsite authorities (including confirmation of the licensee's recommendations to offsite authorities), keeps other Federal agencies informed about the emergency and related NRC actions, and keeps the news media informed of the NRC's knowledge of the emergency. To fulfill this emergency response role, the NRC requires reliable real-time data on plant conditions.

In the past, during an emergency, data on plant conditions was transmitted to the NRC by the licensee through the Emergency Notification System (ENS) (voice communication by telephone). The ENS voice-only emergency communications link required excessive amounts of time for routine transmission of data and for verification or correction of data that appeared questionable. Errors were also encountered in transcribing and interpreting voice transmitted data. Therefore, the Emergency Response Data System (ERDS) was designed to supplement ENS.

Emergency Response Data System (ERDS):

ERDS provides the NRC Operations Center with timely and accurate information from the installed onsite computer systems of nuclear power plants in the event of an emergency at a nuclear power plant. Implementation of ERDS required each licensee (except for Big Rock Point) to establish and maintain a computer program which is designed to transmit a set of 30 selected critical plant parameters. The ERDS will be activated by the licensee upon declaration of an alert or higher emergency condition. Tests with ERDS indicate that a computer-based transmission system is far more accurate and timely than relaying information on plant conditions via telephone.

ERDS will be utilized during (1) emergencies at licensee's facilities, (2) during emergency training exercises, and (3) for periodic testing of the links with the NRC Operations Center. Licensees will activate their ERDS link to begin data transmission as soon as possible (not to exceed one hour) after declaring an alert or higher emergency classification.

ERDS data is available for use by personnel involved in responding to an emergency at the NRC Operations Center, at the Regional Office Incident Response Center, and at the NRC Technical Training Center in Chattanooga, Tennessee. ERDS data is also made available to State governments, upon written request, for their use in emergency response if the State is in the 10 mile Emergency Planning Zone of that plant. The State must provide its own workstation and enter into a Memorandum of Understanding with the NRC governing the use of the ERDS. NRC and the States of Michigan, Washington, Alabama, North Carolina, Arizona, Georgia, Massachusetts, Maryland, New Jersey, New York, Tennessee, Pennsylvania, and Ohio have signed an ERDS MOU and five (5) other States have inquired about similar MOU's with NRC.

Regulations:

ERDS was initiated at licensee facilities on a voluntary basis. The intent throughout this process has been to ensure industry-wide implementation of ERDS. Consequently, in parallel with the voluntary program, rulemaking was initiated to require the implementation of ERDS by all utilities (except for Big Rock Point). On August 13, 1991 the final ERDS rule was published in the Federal Register.

The ERDS rule amended 10 CFR Part 50, requiring licensees to initiate data transmissions to the NRC ERDS computer no later than one hour after the declaration of an Alert, Site Area Emergency, or General Emergency. The licensee is required to provide the necessary software to assemble the data and an output port for each reactor unit in its in-plant computer system. The required emergency data is transmitted to the NRC via an NRC furnished modem over NRC furnished FTS-2000 telephone lines.

The data points to be included in the transmission are those that, to the greatest extent describe specific parameters which are listed in 10 CFR Part 50, Appendix E. Section VI. These parameters are required to be transmitted if they are monitored on plant computer systems, If the data for a selected plant parameter exists, but cannot be transmitted electronically from a licensee's system, then the licensee will continue to provide that data via the existing telephone Emergency Notification System.

Each licensee establishes and maintains a configuration control program which will ensure that the NRC is notified of any changes to the ERDS on-site hardware or software. Any hardware or software changes that affect the transmitted data points identified in the ERDS Data Point Library must be reported to the NRC within 30 days after changes are completed. Any changes that could affect the transmission format and communication protocol to the ERDS must be provided to the NRC at least 30 days prior to the modification.

Current Status:

Twenty-two licensee reactor units successfully implemented ERDS under the voluntary program. The remaining licensees, with the exception of Commanche Peak - Unit 1 and Seabrook, who had schedular exemptions, implemented ERDS prior to the February 13, 1993 deadline. The States of Michigan, Pennsylvania, Maryland, Arizo , Massachusetts, New Jersey, Ohio and Georgia have established the capability to access ERDS data. Ten other States are in the process of implementing links to ERDS.

An operational verification program was imitated on April 5, 1993. Every licensee is required to functionally test their ERDS capabilities on a quarterly basis; States are provided the opportunity to participate in the program. To date, the testing program is proceeding as designed. No significant trend data is yet available.

Contact:

Ken Brockman, Chief, Incident Response Branch
U.S. NRC, Washington, DC 20555. (301) 492-4193

EMERGENCY RESPONSE DATA SYSTEM HIGHLIGHTS

- NRC requires accurate real time data to support its emergency response role.
- ERDS provides direct electronic transmission of preselected plant data from on-site computers to the NRC.
- The system is for emergency use only. Activated by the licensee at or above the alert level of emergency classification.
- Data available to NRC Operations Center, NRC Regional Incident Response Centers, the NRC Technical Training Center, and States within the 10 mile Emergency Planning Zone.
- Data provided includes core and coolant data, containment building data, radioactivity release rates, and meteorological data.
- All plants required by rule (10 CFR Part 50) to implement ERDS prior to February 13, 1993. (Exceptions include Big Rock Point and plants shut down permanently or indefinitely. Schedular exemptions were granted to Commanche Peak - Unit 1 and Seabrook.)
- ERDS supplements the currently installed ENS.

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

Background:

The NRC established the Systematic Assessment of Licensee Performance (SALP) program following the Three Mile Island 2 (TMI-2) accident. Prior to that time, the NRC relied on periodic NRC inspections to identify problems with plant performance. With the SALP process, the NRC developed a routine, systematic approach for the review of all inspection and licensing activities. The NRC uses the SALP process to synthesize its observations of and insights into a licensee's performance and to identify common themes or symptoms. The primary product of the SALP process is the SALP report. The SALP report documents NRC senior management's observations and conclusions regarding licensee performance. The SALP process does not identify nor propose solutions to problems. The licensee is responsible for ensuring plant safety and establishing effective means to measure, monitor, and evaluate the quality of all aspects of plant operations.

The Commission recently approved a number of changes to the SALP program. These changes include: reducing the number of functional areas from seven to four, changing the SALP Board membership to Senior Executive Service (SES) members only, focusing on the assessment of the most significant issues in each functional area, emphasizing recent (within the last 6 months) licensee performance when determining the SALP category ratings, and reducing the report length to promote clearer communications with the licensee and the public. Further, the staff will issue only a single, final, version of the report. The staff implemented these program changes for assessment periods ending after July 19, 1993.

Program Objectives:

The SALP program has four objectives. The first objective is to conduct an integrated assessment of a licensee's safety performance. Senior NRC managers meet at the conclusion of the assessment period to assess licensee performance. A review of the results of NRC inspections and other interactions with the licensee form the basis of the assessment. The second objective of the SALP program is to provide for meaningful dialogue with the licensee regarding safety performance based on the insights gained from the synthesis of NRC observations. The SALP process provides a basis for communications between NRC senior management and licensees regarding plant performance. The third objective of the SALP program is to assist NRC management in making sound decisions regarding allocation of NRC resources used to oversee, inspect, and assess licensee performance. Plants with superior performance are considered for reduced inspection effort while plants with marginal performance will be considered for increased inspection effort. The fourth objective of the SALP program is to provide a method for informing the public of the NRC's assessment of performance. The SALP report is available in the local public document room and the public may attend the SALP meeting with the licensee.

SALP Program Requirements:

The SALP program now has four functional areas, instead of the previous seven, to assess licensee performance. The four functional areas are plant operations, maintenance, engineering, and plant support. Plant support now includes the previous functional areas of radiation protection, emergency preparedness, and security in addition to chemistry, housekeeping and fire protection. Safety Assessment/Quality Verification issues, which were previously considered within a separate functional area, are now considered in all functional areas and addressed in the cover letter, as appropriate. Additional functional areas may be added if the NRC staff believes it is necessary.

The SALP report assigns a category rating for each of the above functional areas, with emphasis on the last six months of performance. The category ratings are as follows: Category 1 is given for a superior level of safety performance; Category 2 is given for a good level of safety performance; and, Category 3 is given for an acceptable level of performance. It should be noted that the lowest SALP rating, Category 3, represents acceptable performance although the margin to unacceptable performance may be small. The NRC does not rely upon the SALP program to identify unacceptable performance.

SALP Process:

The SALP process starts with the assignment of an assessment period. The normal length of a SALP assessment period is 18 months (plus or minus two months). In addition, the NRC regional offices can adjust the length of the SALP period based on plant performance, with shorter assessment periods for plants that need more frequent monitoring. At the conclusion of the SALP assessment period, a SALP Board convenes in the NRC regional office. The SALP Board membership is comprised of four NRC regional and headquarters managers. NRC regional and headquarters staff members responsible for inspection and review activities at the facility make presentations to the Board. The Board assesses performance and assigns a category rating to each of the four functional areas. The SALP report documents the results of the assessment.

The Regional Administrator approves the SALP report and transmits it with a cover letter to the licensee and the public document room. The licensee is given the opportunity to review the report. The NRC and the licensee meet during a special public SALP meeting to discuss the results of the SALP assessment. This SALP management meeting is normally held near the facility.

SALP Results:

The SALP results are utilized by the NRC in the inspection planning process. Those facilities receiving low SALP scores (Category 3) are considered for increased inspection activity, and those facilities with high SALP scores (Category 1) are considered for reduced inspection activity.

Contact:
Gary G. Zech, Chief, Performance and Quality Evaluation Branch
US NRC, Washington, D.C. 20555 (301) 504-1017.

HIGHLIGHTS OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE PROGRAM

- The NRC developed the Systematic Assessment of Licensee Performance (SALP) program following the TMI-2 accident to systematically evaluate observations of and insights into a licensee's performance and to identify common themes or symptoms.

- The SALP program has four objectives: (1) to conduct an integrated assessment of licensee safety performance that focuses on the safety significance of the NRC findings and conclusions during the assessment period, (2) to provide a vehicle for meaningful dialogue with the licensee regarding its safety performance based on the insights gained from synthesis of NRC observations, (3) to assist NRC management in making sound decisions regarding allocation of NRC resources used to oversee, inspect, and assess licensee performance, and (4) to provide a method for informing the public of the NRC's assessment of licensee performance.

- A SALP Board, comprised of four NRC Senior Executive Service managers, convenes to assess licensee performance.

- The SALP Board recommends to the regional administrator a numerical rating of the licensee's performance in each of the four functional areas, with emphasis on the last six months of performance. A Category 1 represents superior performance, Category 2 a good level of performance, and Category 3, acceptable performance.

- The regional administrator assigns the category ratings and issues the SALP report.

- The NRC and licensee conduct a public meeting to discuss the results of the SALP report.

- Superior performance (Category 1) can result in longer SALP assessment periods and reduced inspection effort for the licensee, and acceptable performance (Category 3) can result in shorter SALP assessment periods and increased inspection effort by the NRC.

CHERNOBYL STATUS

Background:

On April 26, 1986, a major accident, determined to have been a reactivity accident, occurred at Unit 4 of the nuclear power station at Chernobyl, Ukraine, in the former USSR. The accident destroyed the reactor and released massive amounts of radioactivity into the environment. After the accident the area in an 18-mile radius around the plant was closed to residential and other access, except for persons requiring official access to the plant and to the immediate area for evaluating and dealing with the consequences of the accident and operation of the undamaged units. The evacuated population numbered approximately 135,000. Thirty-one people died in the accident and its immediate aftermath, most in fighting the fires that ensued; delayed health effects could be extensive, but there are no generally agreed estimates. To stop the fire and prevent a further criticality accident as well as substantial further release of radioactive fission products, boron and sand were dumped on the reactor from the air. The three other units of the four-unit Chernobyl nuclear power station were subsequently restarted. The damaged unit (No. 4) was entombed in a concrete "sarcophagus," to limit further release of radioactive material. Control measures to reduce the radioactive contamination at and near the plant site included cutting down and burying, with the top yard of soil, a pine forest of approximately 1 square mile. The Soviet nuclear power authorities presented a report on the accident at an International Atomic Energy Agency meeting in Vienna, Austria, on August 25-29, 1986.

The Chernobyl reactors are of the RBMK type. These are high-power, pressure-tube reactors, moderated with graphite and cooled with water. Fifteen RBMKs are still in operation in the former USSR. The operating RBMK units incorporate safety improvements made since the Chernobyl accident, though important vulnerabilities remain. U.S. reactors differ in significant ways from the RBMKs.

Status at and Around Chernobyl:

There is concern about the long-term safety of the sarcophagus and long-term dependability of the on-site burial set-up for the buried forest and other massive contaminated materials. Among the sarcophagus concerns is that the faster-than-expected deterioration of structures could lead to rearrangement of materials inside, which could cause an additional release of radioactive material, mainly as dust. The Ukrainian authorities are studying options for dealing with the sarcophagus problem and have been seeking ideas and help from the international community. An international competition in late 1992 - early 1993 produced a number of proposals, mainly from European countries, but there has been no contract award. Design objectives, criteria, and approaches are being studied further by the Ukrainian authorities, with international input.

Studies of longer-term health and environmental consequences of the accident are continuing.

In 1989-90 decisions were taken to evacuate substantial additional nearby areas, in Ukraine and Belarus (and a small area in Russia). About 200,000 people have been relocated so far (including the originally evacuated 135,000). It is not clear at this time to what extent additional relocations will take place.

In an event in October 1991, unrelated to the Chernobyl Unit 4 accident, Unit 2 suffered a fire. There was no significant radioactive material release, but the plant damage was severe. Unit 2 is now shut down; restart is not planned.

Public concern in the areas near Chernobyl, in Ukraine, about the safety of the still operating units led to a Ukrainian government decision to shut them down permanently in 1993. However, in the fall of 1993 that decision was rescinded. The utility (Goskomatom) now plans to run Units 1 and 3 through the late 1990s.

International Studies:

Soviet and successor-state studies of the condition of the sarcophagus, contamination of land and water bodies, health effects, and other post-accident issues continue with international contributions. In the fall of 1989, the First International Workshop on Severe Accidents and Their Consequences, devoted to the Chernobyl accident, was held in Sochi, USSR, under joint sponsorship of the Soviet Nuclear Society and the American Nuclear Society. In mid-1991 an international advisory committee sponsored by the International Atomic Energy Agency completed The International Chernobyl Project: Assessment of Radiological Consequences and Evaluation of Protective Measures. A fact-finding team on the state of the sarcophagus, operating under the auspices of the Organisation for Economic Co-operation and Development's Nuclear Energy Agency, issued its report in October 1992.

Status of USNRC Follow-up:

The U.S. Nuclear Regulatory Commission's review of the Chernobyl accident was divided into three major phases: determining the facts of the accident, assessing the implications of the accident for safety regulation of commercial nuclear power plants in the United States, and conducting specific further studies suggested by that assessment.

The first phase, fact finding, was a coordinated effort among several U.S. Government agencies and some private groups, with the NRC acting as the coordinating agency. The work was essentially completed in January 1987 and updated later that year. The results are reported in NUREG-1250, "Report on the Accident at the Chernobyl Nuclear Power Station."

The second phase, the implications study, was reported in NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," issued after public comment in April 1989. The assessment led to the conclusion that no immediate changes were needed in the NRC's regulations regarding the design or operation of U.S. commercial nuclear reactors.

Plant design, shutdown margin, containment, and operational controls at U.S. reactors protect them against a combination of lapses such as those experienced at Chernobyl. Although the NRC has always acknowledged the possibility of major accidents, its regulatory requirements provide adequate protection, subject to continuing vigilance. The NRC reviews new information that may suggest weaknesses. Assessments in the light of Chernobyl have indicated that the causes of the accident have been adequately dealt with in the design of U.S. commercial reactors.

Yet, the assessment went on to conclude, the Chernobyl accident has lessons for us. The most important lesson is that it reminds us of the continuing importance of safe design in both concept and implementation, of operational controls, of competence and motivation of plant management and operating staff to operate in strict compliance with controls, and of backup features of defense in depth against potential accidents.

Although a large nuclear power plant accident somewhere in the United States is unlikely because of design and operational features, we cannot relax the care and vigilance that have made it so. Accordingly, the assessment led to the recommendation that certain issues should receive further consideration, to provide a basis for confirming or changing existing regulations or staff guidance. Those issues include reactivity accidents, accidents at low power or at zero power (when the reactor is shut down), operator training, and emergency planning.

The Chernobyl follow-up studies for U.S. reactors are the third phase of the NRC review. An overview report on this work, NUREG-1422, "Summary of Chernobyl Followup Research Activities," was issued in June 1992. That report closes out the Chernobyl follow-up research program as such, though certain issues will receive continuing attention in the normal course of NRC work. For example, the long-term lessons with regard to contamination control -- decontamination, ingestion pathway, relocation of people -- will continue to be followed.

Beyond the recommended specific studies, the NRC assessment recognizes that the Chernobyl experience should remain as part of the information to be taken into account when dealing with reactor safety issues in the future.

CONTACT:
George Sege, Technical Assistant to the Director, Office of Nuclear Regulatory Research, U.S.N.R.C., Washington, DC
(301) 492-3904.

CHERNOBYL STATUS
Highlights

- The accident that destroyed Unit 4 of the Chernobyl Nuclear Power Station in the USSR took place on April 26, 1986.
- 31 people died in the accident and its immediate aftermath. Longer-term health effects are being studied.
- 135,000 people were evacuated, from an 18-mile radius exclusion zone around the reactor shortly after the accident. Additional areas are being evacuated. 200,000 people (including the original 135,000) have been relocated to date.
- The damaged Unit 4 reactor is entombed in a "sarcophagus," to prevent further contamination spread. Scientists in Ukraine and other countries are studying the sarcophagus and considering options for dealing with concerns over its long-term stability.
- The NRC, with other U.S. Government agencies and some private groups, conducted a study to determine the facts of the accident. The results were published in 1987. (Report No. NUREG-1250).
- The NRC assessed the implications of the Chernobyl accident for safety regulation of commercial nuclear power plants in the U.S. The results were reported in 1989. (Report No. NUREG-1251). The assessment concluded that no immediate changes were needed in NRC's regulations, because the causes of the accident have been largely anticipated and accommodated in U.S. designs.
- The assessment (NUREG-1251) led to recommendation of certain further studies to provide a basis for confirming or changing regulations. A report on these studies (NUREG-1422) was issued in June 1992.

THREE MILE ISLAND UNIT 2 STATUS

Background:

General Public Utilities Nuclear Corporation (GPUN or the licensee) is in the final phase of the current cleanup effort at Three Mile Island, Unit 2 (TMI-2). Since the March 28, 1979, accident, the licensee has conducted a comprehensive cleanup program designed to place the facility in a safe and stable configuration. Following mitigation of the accident and stabilization of the facility, the major licensee efforts over the past 14 years have included partial facility decontamination, removal of fuel from the reactor vessel and other facilities, offsite shipment of substantial quantities of both high and low level radioactive wastes and the removal, treatment and disposal of the accident-generated water. The licensee has proposed placing the facility into long term storage until Three Mile Island, Unit 1 (TMI-1), also located on the same site as TMI-2, permanently ceases operation, at which time both facilities would be decommissioned. The long term storage period for TMI-2 is called Post-Defueling Monitored Storage (PDMS) by the licensee.

Recent Accomplishments:

On February 1, 1993, GPUN notified the NRC staff that their current best estimate of the residual fuel in the reactor vessel is 2040 pounds (925 kilograms) based on the data from fast neutron measurements. This estimate was derived from calculations made by onsite staff and an independent review by an offsite group headed by Dr. Norman Rasmussen of the Massachusetts Institute of Technology. This estimate was reviewed and confirmed by three additional independent reviewers from national laboratories.

For the balance of the facility external to the reactor vessel, earlier licensee estimates based on measurements, sample analyses, and visual observations indicated that no more than 385 pounds (174.6 kilograms) of residual fuel remains. The NRC staff and their consultants from Battelle Pacific Northwest Laboratories have performed independent evaluations and made independent measurements of these earlier fuel measurements in the auxiliary and reactor buildings. On July 6, 1993, the staff issued an analysis which concluded that the fuel remaining in the TMI-2 reactor vessel will remain subcritical, with an adequate margin of safety, during PDMS.

Evaporation of the treated accident-generated water began in January 1991 after a prolonged period of system testing, modification, and repair. On August 12, _993, the decontamination and evaporation of 2.23 million gallons of accident-generated water was completed.

Post Defueling Monitored Storage (PDMS):

In August of 1988, the licensee submitted a Safety Analysis Report (SAR) to document and support their proposal to amend the TMI-2 license to a possession only license and to allow the facility to enter PDMS. The staff issued final supplement 3 to the Programmatic Environmental Impact Statement for the TMI-2 decontamination and cleanup in August of 1989. In February 1992, the staff issued a Safety Evaluation regarding the PDMS license amendment and a Technical Evaluation Report regarding PDMS. These three NRC staff documents form the basis for the staff position on the acceptability of PDMS. On April 25, 1991, the staff published a notice of opportunity for a prior hearing regarding the licensee request to amend its license. A member of the public petitioned to intervene in the license amendment proceedings. The petitioner, the licensee, and the NRC staff reached a settlement agreement on September 25, 1992. The request to intervene was withdrawn and on October 16, 1992, the Atomic Safety and Licensing Board dismissed the proceeding.

The licensee is currently preparing the facility for entry into PDMS. Preparations for PDMS were completed in October 1992 for the reactor building, and in November 1993 for the auxiliary and fuel handling buildings. The NRC staff and the licensee have prepared a punchlist of remaining items to be completed prior to entry into PDMS. The NRC staff is closely monitoring the licensee progress in satisfying these requirements and commitments. The NRC staff issued the possession only license on September 14, 1993 and expects that TMI-2 will be ready to enter PDMS in late December 1993 or early January 1994.

Decommissioning Funding:

On July 26, 1990, the licensee submitted to the NRC their Decommissioning Funding Plan. The licensee stated in the plan that it will deposit $195 million (1989 dollars) in an escrow account for the radiological decontamination of TMI-2. The value in 1992 dollars is $229 million. The $229 million is in excess of the amount required by NRC regulations for a facility the size of TMI-2. The licensee is required by regulations to submit a preliminary decommissioning plan containing site-specific decommissioning cost estimates five years before the expiration date of the TMI-2 license. Should the preliminary decommissioning plan identify that there are insufficient funds available for decommissioning, the licensee has the remaining five year period to adjust the rate of contributions to the fund to make up any shortfall.

Contact: Lee H. Thonus, DORS/NRR
(717) 948-1164

BP10 (01/94)

HIGHLIGHTS OF THREE MILE ISLAND UNIT 2 STATUS

- Accident occurred at TMI-2 on March 28, 1979.
- Began processing of contaminated water through EPICOR in October 1979.
- Vented 43,000 curies of krypton from the reactor building in July 1980.
- Performed the first manned entry into the reactor building in July 1980.
- Obtained first television pictures of the damaged reactor core in July 1982.
- Removed reactor vessel head and service structure in July 1984.
- Removed the reactor vessel plenum and installed work platform in May 1985.
- Defueling began in October 1985.
- Began core debris offsite shipments in July 1986.
- Completed defueling in April 1990.
- Began evaporation of the accident generated water in January 1991.
- Cost of the cleanup through the end of defueling approximately $980 million.
- Licensee plans to escrow $229 million (1992 dollars) for radiological decommissioning.
- Total occupational exposure 6579 person rem as of the end of calendar year 1992.
- Annual exposure less than 0.1 millirem to the maximally exposed member of the public during defueling.
- Accident-generated water processing completed on August 12, 1993 - total of 2.23 million gallons.
- Possession only license issued on September 14, 1993.

LICENSEE PERFORMANCE INDICATORS

Background:

As a part of its mission to protect the public health and safety, the environment, and national security, the Nuclear Regulatory Commission (NRC) monitors the performance of licensees who operate the 109 commercial nuclear power plants currently in operation in the United States. The findings of this monitoring effort are used by the NRC to adjust its plant-specific regulatory programs. The NRC staff has developed several tools as inputs to the monitoring function. One of these tools is a set of performance indicators. Performance indicators are intended to provide ready information concerning nuclear power plant performance trends and to assist NRC management in identifying poor and/or declining safety performance as well as good and/or improving safety performance.

Performance Indicators:

The Performance Indicator (PI) Program is one aspect of the Commission's efforts to monitor the performance of licensees who operate commercial nuclear power plants in the United States. The program was started in May of 1986, when an interoffice task group began development of the overall NRC program. The first PI Report was published in February of 1987 and contained data on six indicators from the first quarter of 1985 through the fourth quarter of 1986. Reports were provided quarterly to NRC management until June of 1993, when the frequency was changed to twice a year.

Under the direction of the Office for Analysis and Evaluation of Operational Data (AEOD), the PI program has been improved and expanded since it was first introduced. The program currently monitors industry-wide data on eight PIs and evaluates the data to determine performance trends. The eight PIs are: (1) the number of unplanned automatic reactor scrams (trips) while a reactor is critical, (2) the number of selected safety system actuations, (3) the number of significant events, (4) the number of safety system failures, (5) the forced outage rate, (6) the number of equipment-forced outages per 1000 commercial critical hours, (7) the collective radiation exposure, and (8) cause codes. The AEOD staff provides biannual reports containing plant-specific data for these eight PIs to the Commission and to NRC senior managers. The reports are also placed in the NRC Public Document Room. In addition, the staff provides plant-specific information and industry average data extracted from each PI report to licensee managers.

The PI Program is a single, coordinated, overall NRC program that provides an additional view of operational performance and enhances the NRC's ability to recognize areas of changing safety performance of operating plants. However, it is only a tool that must be used in conjunction with other tools, such as the results of routine and special inspections and the Systematic Assessment of Licensee Performance (SALP), to provide data to NRC managers who must decide whether any plant-specific regulatory programs need adjusting. It should be recognized that PIs have limitations and are subject to misinterpretation. Performance indicators alone do not provide a basis for ranking individual power plants or taking regulatory actions and are not used in communications with licensees as a measure of performance level. The NRC has a sensitivity to pressure which would cause licensee personnel at individual power plants to "manage the indicators" or take any actions that are contrary to plant safety because of performance indicators. The PIs for a given plant, when viewed as a set, provide additional data for assessing changes in plant operational performance. The PIs focus attention on the need to assess and understand underlying causes of identified changes by evaluating other available information.

The NRC will continue its review, evaluation, and revision, as needed, of the PI Program. Development and implementation of risk-based indicators is on-going and will continue. Further program revisions will be made as appropriate.

Contact:

Donald E. Hickman, Chief, Performance Indicator Section,
US NRC, Washington, DC 20555. Phone: (301) 492-4431

HIGHLIGHTS OF LICENSEE PERFORMANCE INDICATORS

- NRC monitors licensee performance to adjust plant-specific regulatory programs
- One monitoring tool is a set of performance indicators (PIs) to identify performance trends
- Quarterly reports provided from 1987 to June 1993, twice a year after that
- Eight indicators evaluated to determine performance trends: (1) automatic scrams while critical, (2) selected safety system actuations, (3) significant events, (4) safety system failures, (5) forced outage rate, (6) equipment forced outages per 1000 commercial critical hours, (7) collective radiation exposure, and (8) cause codes
- Provided to the Commission, NRC senior managers, licensee senior managers, and the public
- PIs have limitations, are subject to misinterpretation, and must be used with other tools to assess regulatory programs
- Not to be used alone for ranking plants, taking regulatory actions, or as a measure of performance level
- Used as a set to focus attention on the need to assess and understand underlying causes
- Review, evaluation, and revision of the PI Program, including risk-based indicators, is on-going

NUCLEAR REACTOR RISK TO THE PUBLIC (ACCIDENTS)

Background:

The safe operation of a nuclear power plant requires the consideration of various types of possible accidents which may pose a risk to the public. The accidents range in terms of severity and likelihood of occurrence. Risk is considered to be the combination of accident severity (consequences) and the likelihood (frequency) of the accident. The risk due to operation of nuclear power plants can never be zero, just as the risks we face from other sources such as illness and auto accidents can never be zero. Regulatory requirements and attention are necessary to assure that the risk from nuclear power plant operation is very low when compared to all other types of risk that we face every day. The Nuclear Regulatory Commission's (NRC's) responsibility is centered around the application and enforcement of the applicable regulatory requirements as described in Title 10 of the Code of Federal Regulations. The intent is to assure that the risks are maintained at acceptably low levels.

Nuclear Reactor Accident Risks:

Nuclear power plants are designed to confine the fission products which accumulate within the nuclear fuel. Part of the overall risk stems from the possibility of accidental release of the fission products into the environment beyond the plant site boundaries. To ensure that this risk is kept at acceptable levels, the concept of "defense-in-depth" is applied to the design, licensing, and operation of nuclear power facilities.

For example, the physical confinement of fission products is implemented by way of multiple barriers such as fuel cladding, reactor coolant system vessel and piping, and a containment building. The need to maintain the integrity of the reactor core (fuel) and avoid damage requires that an adequate supply of water be provided for cooling it. Here again, "defense-in-depth" is implemented by providing diverse and multiple backup systems so that there is an adequate assurance of a supply of water for cooling the core. Due consideration is given to keeping the plant within safe operating limits and conditions (technical specifications). Other safety measures include paying attention to the availability and reliability of plant equipment, plant maintenance, operator training, and plant management in order to minimize the overall risk.

Regulatory reviews, analyses and inspections are used to ensure that these measures comply with appropriate NRC regulations and that the estimated risk is acceptably low.

Regulations:

Title 10 of the Federal Code of Regulations (10 CFR) contains the NRC's criteria and requirements for ensuring an acceptable level of safety with respect to nuclear power plants. As an adjunct to the regulations, the NRC has developed a set of Regulatory Guides and the Standard Review Plan in order to clarify the regulatory requirements. The NRC also issues various generic communications that address safety related concerns.

None of the regulations relate directly to quantitative risk measures. The regulations are viewed in terms of the use of sound engineering concepts to provide what is judged to be an acceptable level of safety. However, the NRC has issued a policy statement that describes safety goals for the operation of nuclear power plants.

The qualitative safety goals are that nuclear power should pose no significant additional risk to the life and health of individuals and that societal risk should be comparable to other viable competing energy supply technologies with no significant addition to other societal risks.

The quantitative safety goals are that the risk of an individual prompt fatality should not exceed 0.1% of that due to other accidents and that the risk of cancer fatalities to nearby population should not exceed 0.1% of the total cancer risk from all other causes.

Current Status:

Currently, 109 commercial nuclear power plants are licensed to operate in the U.S. The NRC believes that by meeting the existing regulations these plants pose an acceptably low level of risk to the public. However, in 1988, the NRC issued a requirement for the utilities licensed to operate nuclear power plants to perform a plant-specific search for vulnerabilities to severe accidents. In this effort, known as the Individual Plant Examination (IPE) program, virtually all licensees have chosen to use probabilistic risk assessment methods. This program is directed towards the identification of possible plant weaknesses with respect to safety that stem from considerations beyond those covered by the existing regulations. The intent is to ensure that current risk is as low as previously believed. The first phase of this program is devoted to investigating accidents which could be initiated within nuclear plants and is expected to take about three years to complete. Sixty-three IPEs have been submitted to the NRC to date. Weaknesses identified in this process are being addressed by licensees and reviewed by the staff. The second phase is addressing accidents which could be initiated externally, such as earthquakes and floods. The second phase was started in 1990 and is expected to be completed in 1995.

In order to improve the means for evaluating risks associated with the operation of nuclear power plants, the NRC is continuing to improve and refine probabilistic risk assessment (PRA) methods. At the same time, the NRC is monitoring plant operations, equipment failures, operator errors and similar plant performance features in order to identify potential problems before they become serious.

Contact:

Mark P. Rubin, Acting Chief, Probabilistic Safety Assessment Branch US NRC, Washington, D.C. 20555 (301) 504-1089

HIGHLIGHTS OF NUCLEAR REACTOR RISK TO THE PUBLIC

- As a result of the corrective steps taken by the NRC and the nuclear power industry since the Three Mile Island accident, the estimated probability of a nuclear reactor accident has decreased significantly.
- The NRC has issued a policy statement describing the safety goals with respect to nuclear power plants.
- The risks associated with the operation of nuclear power plants that meet existing safety regulations are acceptably low.
- The NRC has implemented the Individual Plant Examination program in order to identify any weaknesses to events beyond the design basis of the plant that may exist and to ensure that low risk levels are maintained.

ADVANCED REACTORS PROGRAM

The NRC's "Statement of Policy for Regulation of Advanced Nuclear Power Plants (Final Statement)", July 8, 1986, encourages early interaction (prior to license application) between NRC and advanced reactor designers to provide licensing guidance applicable to those designs. In June 1988 the NRC issued NUREG-1226, a report on "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants". The report provides guidance on the implementation of the policy, and describes the staff approach to be used in the review of advanced reactor concepts under the policy statement. Other important Commission guidance to be used in the early interaction process with sponsors of advanced designs include the Severe Accident Policy Statement (50 FR 32138; August 8, 1985), the Standardization Policy Statement (52 FR 34884; September 15, 1987), and the Safety Goal Policy Statement (51 FR 30028; August 21, 1986).

Thus the staff is prepared to conduct early, interactive reviews of advanced reactor designs including consideration of the Commission's regulations, regulatory guides and other guidelines. These include such established and developing criteria as the defense-in-depth philosophy, standardization, the Commission's safety goal and severe accident policies, and applicable industry codes and standards. NRC early interaction with a potential applicant is in the context of a preapplication review which takes about two years and culminates in a preapplication safety evaluation report. The objectives of the preapplication review are (1) to identify major safety issues that could require Commission policy guidance to the staff, (2) to identify major technical issues that the staff could resolve in the context of existing regulations or Commission policy, and (3) to identify research and development needed to resolve identified issues.

The early reviews discussed here are done prior to formal submittal of an application for a standard design certification pursuant to 10 CFR Part 52 (54 FR 15372; April 18, 1989). This regulation was enacted by the Commission to provide a stable procedural framework for the early resolution of licensing issues and finally, certification of a standardized design by rulemaking.

CANDU 3

The CANDU 3 design is a single-loop pressurized heavy water reactor rated at 450 MWe with two steam generators and two heat transport pumps connected in series. The design utilizes natural uranium fuel, separate heavy water moderator and reactor coolant, computer-controlled operation, and on-line refueling. The reactor has 232 horizontal pressure tubes supported in a calandria tank filled with the heavy water moderator. The tank also supports the reactivity regulating and safety devices which are inserted between and among the pressure tubes. The CANDU 3 design is an evolution of the CANDU 6 design (600 - 640 MWe). The CANDU 6 has been approved by the Atomic Energy Control Board (AECB), the Canadian government agency responsible for regulating atomic energy in Canada, for operation in Canada. Four CANDU 6 units are currently in operation (two in Canada, one each in Korea and Argentina) and 28 other CANDU-type units have been built around the world, but none in the U.S. The CANDU 3 contains many features and components already in use in the CANDU 6 design.

The CANDU 3 design is being developed by the Atomic Energy of Canada, Limited (AECL), whose design facilities are based in Mississauga, Ontario, and Saskatoon, Saskatchewan, Canada. AECL Technologies, a United States subsidiary of AECL, Incorporated, informed the NRC of its intent to seek design certification of the CANDU 3 design under the provisions of 10 CFR Part 52 in a letter to then-Chairman Lando Zech, dated May 25, 1989. AECL Technologies subsequently submitted, and the NRC is reviewing, a two volume Technical Description and a two volume Conceptual Safety Report describing the CANDU 3. In addition, AECL Technologies has submitted a number of technology transfer reports describing various aspects of the CANDU technology.

The NRC is continuing activities to prepare for a CANDU 3 design certification application by becoming familiar with the design, by maintaining technical progress on key issues, and by working on code development and benchmarking. In a letter dated August 3, 1993, AECL Technologies stated their intent to file a Standard Design Certification application within about one year. The staff is identifying the resources necessary for carrying out a standard design certification review of the CANDU 3 design.

PIUS

PIUS (Process Inherent Ultimate Safety) is a 640 MWe advanced pressurized water reactor (PWR) design, by ABB Atom of Sweden, that utilizes natural physical phenomena to accomplish control and safety functions usually performed by electromechanical devices. The PIUS design consists of a vertical pipe, called a reactor module, which contains the reactor core and is submerged in a large pool of highly borated water. The reactor core is comprised of fuel elements that are similar to current PWR elements. The borated pool water is provided to shutdown the reactor and to cool the core by natural circulation while the reactor is shutdown. The reactor module is open to the borated pool at the bottom and again at the top of the reactor module. Devices referred to as density locks are provided at these two openings. During normal operation at design power, the density locks prevent the entry of the highly borated pool water into the reactor module due to the pressure balance maintained across the tube bundle. In normal operation, the primary loop reactor water enters the reactor module from the steam generator, flows up through the core, out of the top of the reactor module to the steam generator, and is pumped back into the bottom of the reactor module. Under certain transient conditions, the pressure balance across the density locks is not maintained and the borated pool water flows into the core and shuts down the reactor. Unlike most reactors, PIUS does not use control rods for regulating reactivity. Reactivity is controlled by the boron concentration and temperature of the primary loop reactor water.

The steam generating equipment of the PIUS design is similar to that of a typical U.S. or European pressurized light water reactor plant. One important difference in plant design is the very large, by current standards, prestressed concrete reactor vessel. This vessel holds both the reactor module and the borated pool.

In October 1989, ABB Atom requested the NRC to perform a licensability review of their PIUS Preliminary Safety Information Document (PSID). However, since then, the staff and ABB-CE have agreed that further work on the preapplication review of the PIUS PSID would not be meaningful given the limited resources available. The staff is in the process of documenting the review of the PIUS design by March 1994 and will terminate all other activities until an application for design is submitted by ABB-CE. Presently there is no date for a design certification application.

MHTGR

The modular high temperature gas-cooled reactor (MHTGR) design is a helium-cooled and graphite-moderated thermal power reactor. The fuel is millions of ceramic coated microspheres distributed in cylindrical rods which are inserted in large hexagonal graphite blocks. The blocks are stacked vertically within the reactor vessel through which the pressurized helium coolant is circulated. The plant design consists of four identical reactor modules, each with a thermal output of 350 MWt, which are coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MWe, with a power conversion of approximately 40%. This is about half the power output of current single light water reactor units. The design includes passive reactor shutdown and decay heat removal features to minimize required reactor operator actions.

The NRC has licensed one high temperature gas reactor, Fort St. Vrain in Colorado, which was permanently shutdown in August 1989. The advanced MHTGR design, sponsored by the Department of Energy (DOE), has been under review at the NRC since 1986, and a draft Preapplication Safety Evaluation Report (PSER) was issued in

March 1989 (NUREG-1338). Important safety matters are fuel design and performance, containment design and performance, reactor cavity cooling system, accident selection and analysis, accident source terms and analysis, role of the operators, design of the control room and remote shutdown area, emergency preparedness, and quality standards for equipment. The current review effort to update the draft report and issue a final PSER is tentatively scheduled for completion by the Spring of 1995.

PRISM

The design submitted by the Department of Energy (DOE) is a small, modular, pool-type, liquid-sodium cooled reactor producing 471 MWt. The reactor fuel elements are cylindrical tubes containing pellets of uranium-plutonium-zirconium metal alloy. The reactor size was selected to permit use of passive shutdown and decay heat removal features.

The PRISM standard plant is to consist of nine reactor modules arranged in power blocks of three reactor modules with their steam generators supplying steam to one turbine-generator. The power output of the standard site would be 1395 MWe.

Each of the reactor modules would be located in a silo below grade. The steam generator and secondary system hardware would be located in a separate building and would be connected by a below-grade pipeway. The reactor modules would share a common control center, nuclear island maintenance building, and reactor service building.

In general, the PRISM design features have been chosen to prevent core damage events that previous Liquid Metal Reactor (LMR) designs have traditionally been designed to accommodate.

DOE submitted the conceptual design to NRC for preapplication review in November 1986. The NRC published a draft PSER in September 1989 (NUREG-1368). The staff's preapplication review is to provide guidance to the design's sponsors early in the design process. The current review effort is directed at updating the draft PSER and publication by a final PSER is tentatively scheduled for the end of January 1994.

Contact:

Richard W. Borchardt, Project Director, Advanced Reactors Project Directorate, US NRC, Washington, DC 20555, (301) 504-1118

HIGHLIGHTS - ALL PROJECTS

CANDU 3 --

- AECL Technologies, a division of AECL, Incorporated, submitted a letter in May 1989 stating intent to seek design certification of the CANDU 3 pursuant to 10 CFR Part 52.
- A number of documents were submitted to NRC for preapplication review between mid-1990 and early 1991.
- Several meetings have been held among the NRC, AECB, AECL, and AECL Technologies to define the scope of the preapplication review and to discuss preliminary technical issues identified by the NRC.
- In an August 3, 1993, letter, AECL Technologies stated their intent to submit the CANDU 3 design for standard design certification in about one year.
- The staff preparation for design certification review is ongoing.

PIUS --

- ABB requested a licensability review of the PIUS design in October 1989.
- ABB submitted the PIUS Preliminary Safety Information Document (PSID) for review in May 1990.
- First review of PIUS began in June 1991.
- Status of the review to be documented by March 1994. All other activities stopped until a design certification application is received.

MHTGR --

- Preliminary Safety Information Document (PSID) submitted by DOE in 1986.
- Draft Preapplication Safety Evaluation Report issued by NRC in March 1989.
- DOE submitted PSID Amendments 11 through 13 responding to issues, December 1991, and March and August, 1992.
- NRC tentative schedule for final PSER for the MHTGR design is Spring 1995.

PRISM --

- Preliminary Safety Information Document (PSID) submitted to NRC in November 1986.
- NRC Draft Preapplication Safety Evaluation Report (PSER) on PRISM (NUREG-1368) released September 1989.
- DOE submitted Amendments 12 and 13 responding to issues, March and May 1990.
- NRC schedule for final PRISM Preapplication Safety Evaluation Report (PSER) is January 1994.

HIGH-LEVEL RADIOACTIVE WASTE BRIEFING UPDATE

Background:

High-level radioactive waste (HLW) means: (1) irradiated (spent) reactor fuel, (2) liquid wastes resulting from the operation of the first cycle solvent extraction system, and the concentrated wastes from subsequent extraction cycles, in a facility for reprocessing irradiated reactor fuel, and (3) solids into which such liquid wastes have been converted. HLW is primarily in the form of spent fuel from commercial nuclear power plants; it also includes some reprocessed HLW from defense activities, and a small quantity of reprocessed commercial HLW. Current plans for management of HLW call for the development of a monitored retrieval storage (MRS) facility by 1998, and a permanent HLW repository deep beneath the surface of the earth by the year 2010. The U.S. Department of Energy (DOE) has the responsibility for disposing of HLW. The U.S. Environmental Protection Agency (EPA) is responsible for developing appropriate environmental standards for HLW. The Nuclear Regulatory Commission has the licensing authority for the disposal and long term storage of HLW.

High-Level Radioactive Waste:

This country's policies governing the permanent disposal of HLW are defined by the Nuclear Waste Policy Act of 1982 (NWPA) and the Nuclear Waste Policy Amendments Act (NWPAA) of 1987. To provide the long-term permanent isolation required, the NWPA specifies that HLW will be placed in deep-underground geologic repositories to be built and operated by DOE. To this end, DOE is developing a waste management system consisting, in part, of a geologic repository in which HLW can be permanently isolated deep beneath the surface of the earth, and an MRS in which waste can be stored prior to permanent disposal. NRC has the licensing and related regulatory authority for both the MRS and HLW geologic repository.

An MRS facility is an integral part of the waste management system being proposed by DOE for achieving timely acceptance of spent fuel. NWPAA allows a dual approach to MRS siting: (1) siting by DOE, through a process of survey and evaluation; and (2) siting through the efforts of the Nuclear Waste Negotiator.

Through the NWPAA, Congress designated the Yucca Mountain site in Nevada as the single candidate site for characterization as a potential geologic repository. The Yucca Mountain site has not been selected for a repository; rather, it has been chosen as the only site to be characterized at this time.

Site characterization is a program of exploration and research, both in the laboratory and in the field, undertaken to establish the geologic conditions and the ranges of those parameters at a particular site. Site characterization includes borings, surface excavations, excavation of exploratory shafts or ramps, subsurface lateral excavations and borings, and in situ testing at depth to determine the suitability of the site for a geologic repository.

Regulations:

The NRC's requirements governing the disposal of HLW in a geologic repository are contained in Title 10 Code of Federal Regulations, Part 60 (10 CFR Part 60). These regulations govern prelicensing activities, authorization for DOE to begin construction of the facility, authorization for DOE to receive and place the wastes in the facility, and authorization for DOE to close the facility (license termination).

The NRC's requirements governing the storage of HLW in an MRS facility are contained in Title 10 Code of Federal Regulations, Part 72 (10 CFR Part 72). These regulations establish requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel and other radioactive material associated with spent fuel storage.

The EPA's standards for the disposal of HLW in a geologic repository are contained in Title 40 Code of Federal Regulations Part 191 (40 CFR Part 191). These regulations establish generally applicable environmental standards for the management and disposal of spent nuclear fuel and other HLW. The NRC is responsible for implementing these standards in 10 CFR Part 60.

Current Status:

Currently, the repository program is focused on prelicensing site characterization activities. In the prelicensing phase, one of NRC's primary responsibilities is to review DOE's site characterization plan and associated activities, and to provide comments to DOE identifying any specific concerns. In addition, NRC staff observes various site characterization activities in the field, such as drilling and tunneling, and also observes DOE quality assurance surveillances and audits. All prelicensing consultation activities are open to participation by the State of Nevada, affected Indian Tribes, and affected units of local governments.

On June 3, 1993, NRC and DOE signed the revised Procedural Agreement and the Project-Specific Agreement. The Procedural Agreement was originally signed in 1983, and the Project-Specific Agreement was originally signed in 1984. Together, these agreements provide the basis for implementing most of the NRC/DOE interactions. The substantive changes that were incorporated into the Procedural Agreement dealt with newly added or revised guidelines for conducting technical exchanges, site visits, licensing and management meetings, and quality assurance audits and surveillances. Similarly, major changes in the Project-Specific Agreement included revised guidelines for preparing interaction reports, maintaining and distributing site characterization data, communications between points of contact from NRC and DOE project offices, acquisition of samples by NRC contractors from DOE during site characterization activities, and specific NRC On-site Licensing Representative responsibilities and authority.

DOE completed its site characterization plan for the Yucca Mountain site in December 1988. The NRC staff completed its review of that document in July 1989, and concluded that overall, it was a usable plan for site characterization. Originally, the staff identified two objections to DOE starting site characterization. One objection concerned the DOE quality assurance (QA) program, and the other was related to the design process for the Exploratory Studies Facility (ESF). Additionally, 196 other concerns in the form of comments and questions were raised.

Regarding the QA objection, NRC notified DOE by letter, dated March 2, 1992, that the objection was removed. NRC determined that all organizations participating in site characterization activities had developed, and are implementing, a QA program that meets NRC requirements. NRC continues to monitor QA program implementation through audits and surveillances.

DOE provided information in response to the ESF objection on March 3, 1992. Based on that information, and information contained in two DOE reports ("Exploratory Studies Facility Alternatives Study: Final Report" and "Risk/Benefit Analysis of Alternative Strategies for Characterizing the Calico Hills Unit at Yucca Mountain,") and observations of DOE design reviews, the NRC staff determined that the ESF objection should be lifted. In a letter dated November 2, 1992, NRC notified DOE that this objection was lifted. The staff continues to track DOE's activities related to the ESF objection and implementation of the ESF design control process through participation in ESF design reviews, reviews of site characterization Progress Reports, participation in DOE audits of the ESF design review process, and participation in bi-monthly meetings with DOE to discuss ESF design, design control process and technical inputs to the design.

In 1991, the State of Nevada granted DOE the permits necessary for DOE to proceed with surface based site characterization activities. These activities include the excavation of test pits and trenches, borehole drilling, and hydrologic monitoring to address technical issues related to volcanism, radionuclide transport, seismicity, and faulting. DOE continues to actively conduct site characterization field work in these areas at the Yucca Mountain Project Site and vicinity.

On November 30, 1992, DOE began work on the site access road to the main portal for the ESF at Yucca Mountain, and work on the portal pad began in April 1993. Initial work on the underground opening used the drill and blast method to excavate a starter tunnel, of approximately 200 feet in length, for the tunnel boring machine (TBM), which will be used to excavate the main ramp-tunnel of the ESF (about 25,000 feet). Excavation of the starter tunnel began in April 1993 and was completed in September 1993. The TBM will be delivered to the site in spring 1994 and DOE proposes to begin excavation of the north ramp portion of the main ramp-tunnel of the ESF in August 1994.

In a letter of August 20, 1993, NRC notified DOE of concerns related to the design and design control process based on its observations and reviews of DOE activities. The letter cites deficiencies identified by DOE QA and technical personnel during recent QA audits of DOE's Civilian Radioactive Waste Management System Management and Operating Contractor (M&O) and requests that DOE provide a rationale for proceeding with activities related to design and construction while the deficiencies are being investigated and corrected. The deficiencies include inadequate procedures, failure to follow procedures, and inadequate documentation of design bases. In addition, it is not clear that technical information, identified in DOE reports as necessary for ESF design decisions, will be collected in time to provide input to the ESF design. Other items requested in the letter include an action plan for corrective actions for the M&O design deficiencies, a controlled baseline ESF design integrated with a geologic repository operations area conceptual design, and a detailed plan for the process DOE will use to keep the staff informed of future design changes. DOE submitted a formal response to NRC's letter on November 18, 1993, that is currently under review by the NRC staff. In addition, DOE now provides design review information to the staff prior to all design reviews and meets with the staff to discuss ESF design status and concerns on a bi-monthly basis. DOE has also scheduled an audit of the M&O's design activities in April 1994. That audit will be observed by NRC QA staff.

The NWPA calls for a 1998 date for DOE to accept spent fuel from utilities. Until recently, the DOE approach has been to use the Nuclear Waste Negotiator for the voluntary siting of an MRS that would begin acceptance of spent nuclear fuel by 1998. The Office of the Nuclear Waste Negotiator was established by the NWPAA to find a state or Indian Tribe willing to host a repository or MRS at a technically qualified site. Interest has been expressed by nine groups in evaluating the feasibility of hosting an MRS

(Phase IIA). To date, two of these groups have gone on to Phase IIB where work will focus on detailed environmental and site feasibility studies. In addition, DOE has recently begun studying the feasibility of locating an MRS on federal land. DOE's efforts are proceeding in parallel with those of the Nuclear Waste Negotiator in order to meet the 1998 objective established in the NWPA.

In October 1992, DOE initiated a study to evaluate the feasibility of using multi-purpose canisters (MPC) in the waste management system. The MPC concept is to use a common container that has different overpacks for transportation, storage, and disposal. The purpose of the MPC is to create first, a compatible approach for the transportation and storage of spent nuclear fuel, and then consider compatibility with final disposal. DOE completed its MPC study and held workshops in July and November 1993 to obtain input from interested parties in developing the MPC concept. DOE expects to issue a request for proposals for MPC designs in spring 1994.

EPA developed generally applicable environmental standards for a HLW repository that were promulgated as 40 CFR Part 191 in 1985. These standards were remanded in 1987 due to inconsistencies with other EPA standards with respect to individual dose and ground-water protection. Since that time EPA has been working on revising its standard. However, in late 1992, Congress passed the Energy Policy Act (EnPA) of 1992 which required EPA to contract with the National Academy of Sciences (NAS) to conduct a study on specific aspects of these standards and issue findings and recommendations. The NAS plans to complete this study by the end of 1994. Among the issues to be included in the Academy's review are: (1) the reasonableness of a health-based standard based on individual dose; (2) the ability of post-closure oversight to prevent an unreasonable risk of breaching the repository's barriers or increasing the exposure of the public to radiation beyond allowable limits; and (3) the capability to make scientifically supportable predictions of the probability of human intrusion for 10,000 years.

The EnPA further requires that the EPA promulgate by rule, public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site, based upon and consistent with the findings and recommendations of the NAS. These standards are to be promulgated not later than one year after EPA receives the findings and recommendations of NAS. Furthermore, the EnPA also requires that NRC, within one year of the promulgation of the EPA standards, amend its technical requirements and criteria to conform with these standards.

CONTACT:
C. William Reamer, Acting Director
Repository Licensing & Quality
Assurance Project Directorate
Division of High-Level Waste Management
(301) 504-3387

HIGHLIGHTS OF HIGH-LEVEL RADIOACTIVE WASTE (HLW)

- NWPA (1982) and NWPAA (1987) lay out a national program for disposal of HLW in a deep geologic repository and possible interim storage in an MRS
- NWPAA designated Yucca Mountain, Nevada, for characterization as a potential repository site
- NRC requirements for the interim storage of HLW are contained in 10 CFR Part 72
- NRC requirements for the disposal of HLW are contained in 10 CFR Part 60
- EPA standards for the disposal of HLW are contained in 40 CFR Part 191
- The EnPA requires that: NAS conduct a study on specific aspects of the HLW environmental standards and make recommendations on reasonable standards by December 31, 1993; EPA revise its standards based upon, and consistent with, the NAS findings and recommendations within one year after it receives the NAS findings; and NRC amend its technical requirements and criteria to conform to the EPA standard within one year of the promulgation of EPA's standards
- NRC is currently involved in prelicensing interactions and review of DOE HLW repository site characterization activities
- NRC is currently involved in prelicensing interactions and review of DOE MRS activities
- DOE to submit to NRC a HLW repository license application for construction authorization in 2001
- DOE to begin waste emplacement in a HLW repository in 2010
- All prelicensing consultation activities are open to participation by the State of Nevada, affected Indian Tribes, and units of affected local governments

REMEDIATION OF RESIDUAL RADIOACTIVITY AT NON-ROUTINE CONTAMINATED SITES

Background:

There are about 50 sites contaminated with radioactive material, under the jurisdiction of the Nuclear Regulatory Commission, throughout the country, that are considered to be non-routine decommissioning cases. To compel timely remediation of these sites, NRC initiated the Site Decommissioning Management Plan (SDMP) in 1990. NRC emphasis on timely remediation of the sites resulted from former Chairman Carr's August 3, 1989, testimony before the Subcommittee on Environment, Energy, and National Resources, chaired by Congressman Synar, and from continuing NRC concern about lack of progress at some of these sites.

Discussion:

The NRC staff developed the SDMP to identify sites requiring remediation and to provide the Commission with a status report on the actions taken to bring about the remediation of the SDMP sites. In summary, the SDMP contains the following information:

1) definition of the project management plan;

2) identification of the sites requiring decommissioning;

3) schedule and resources necessary to support NRC actions to regulate the remediation of the contaminated sites;

4) resolution of policy and Congressional issues for SDMP implementation and minimizing problems with future contaminated sites.

The SDMP not only identifies the sites requiring decommissioning, but also describes:

1) the site;

2) the wastes and radioactivity remaining on the site;

3) the radiological hazard from the remaining wastes;

4) the financial assurance required;

5) the status of the remediation activities; and

6) NRC proposed actions and schedules to ensure timely decommissioning.

SDMP Criteria:

A site is listed in the SDMP if it meets one or more of the following criteria:

1) problems with a viable responsible organization (e.g., inability to pay for or unwillingness to perform decommissioning);

2) presence of large amounts of soil contamination or unused settling ponds or burial grounds that may be difficult to dispose of:

3) long-term presence of contaminated, unused facility building;

4) license previously terminated; or

5) contamination or potential contamination of the groundwater from onsite wastes.

Regulations:

In 1988, NRC promulgated the final decommissioning regulations, under 10 CFR Parts 30, 40, 50, 70, and 72. The regulation defines decommissioning as removing a facility from service, reducing the residual radioactivity to a level that will permit release of the facility for unrestricted use, and termination of the license. In summary, the decommissioning regulations prescribes requirements for decommissioning planning, financial assurance, recordkeeping, and license termination.

SDMP Action Plan:

The SDMP has been effective in ensuring coordination and resolution of some policy and regulatory issues affecting site decommissioning. Progress, from 1990 through 1992, on actual site remediation, however, was slow. Because of that limited progress, the staff developed the SDMP Action Plan, which was approved by the Commission on April 6, 1992. The Action Plan:

1) identifies current criteria to guide remediation of contaminated soils, structures, and equipment and emphasizes site-specific application of the as low as reasonably achievable (ALARA) principle;

2) states NRC's position on the finality of decommissioning decisions;

3) describes NRC's general expectation that SDMP site remediation will be completed within a 4-year timeframe after operations cease for 3-years after the issuance of an initial decommissioning order;

4) identifies currently available guidance on site characterization work in support of decommissioning; and

5) describes the process the staff will use to establish and enforce schedules for timely decommissioning on a site-specific basis.

The Action Plan has been effective in communicating, to licensees and the public, the Commission's expectation that SDMP sites will be remediated in a timely and effective manner. Since the release of the Action Plan, licensees have been generally cooperative in establishing reasonable, but firm, decommissioning schedules, and initiating remediation activities, without the need for enforcement action.

Policy Issues Requiring Resolution:

The SDMP contains a series of policy issues, related to the remediation of contaminated materials licensee sites, that need to be resolved. Resolution of these policy issues will provide a regulatory framework for more consistent and efficient licensing actions related to site decommissioning in the future. Two issues require prompt resolution for more effective implementation of the SDMP: 1) developing a national standard for residual radiological contamination that is acceptable for releasing a site or materials for unrestricted use; and 2) developing a rule to require timely decommissioning.
The Commission recently initiated an enhanced participatory rulemaking process to establish radiological criteria for decommissioning. Representatives of State governments, Tribal governments, local governments, other Federal agencies, environmental groups, citizen groups, and industry groups were invited to participate in the workshops associated with the rulemaking. The workshops began in January 1993 and ended on May 7, 1993. A proposed rule is scheduled to be published in the Federal Register in June 1994. Final requirements are scheduled to be promulgated in the May 1995.

A proposed rulemaking on the timeliness of decommissioning was published in the Federal Register on January 13, 1993. The final rule is scheduled to be promulgated in March 1994.

Current Status:

The NRC staff is implementing the SDMP as described in SECY-93-179 dated June 24, 1993. The SDMP was published in October 1993 as NUREG-1444. The decommissioning of the Allied Signal Aerospace - Bendix Division site, in Teterboro, NJ, was completed, and removed from the SDMP list on February 28, 1992. Decommissioning of the Budd Company site, in Philadelphia, PA, was completed in late 1992, the license was terminated in April 1993, and the site was removed from the SDMP in April 1993. Decommissioning actions at the Old Vic site in Cleveland, OH, have been completed and the site is being removed from the SDMP list. Two other sites, Amax, Incorporation, in Wood County, WV, and UNC Recovery Systems, in Wood River Junction, RI, have completed site remediation activities. The licenses for these two sites will be terminated, and the sites removed from the SDMP list, after pending administrative or jurisdictional issues are resolved. In addition, decommissioning actions at the Texas Instruments site, in Attleboro, MA, and the Nuclear Lake site, in Pawling, NY, have been completed, and removal from the SDMP list is expected to be completed in early 1994. Attached is the current list of SDMP sites.

Contact:
John H. Austin, Chief, Decommissioning and Regulatory Issues Branch, Division of Low-Level Waste Management and Decommissioning, Office of Nuclear Material Safety and Safeguards, 301-504-2560.

HIGHLIGHTS

- About 50 sites contaminated with radioactive material are non-routine decommissioning cases.

- Site Decommissioning Management Plan (SDMP) developed in 1990.

- SDMP includes information on identification of sites, schedule, and resources to support NRC actions.

- Decommissioning rule promulgated in 1988 in Parts 30, 40, 50, 70, and 72.

- Decommissioning defined as removing a facility from service, reducing the residual radioactivity to a level that will permit release of the facility for unrestricted use, and termination of the license.

- SDMP Action Plan approved by the Commission on April 6, 1992.

- Policy issues that require resolution:

  1) development of residual contamination criteria,
  2) timeliness of decommissioning.

CONTAMINATED SITE LIST[1]

Advanced Medical Systems, Inc.; Cleveland, OH
Aluminum Company of America; Cleveland, OH
Amax; Wood County, WV
Anne Arundel County/Curtis Bay; Anne Arundel County, MD
Army, Department of, Aberdeen Proving Ground; Aberdeen, MD
Babcock & Wilcox; Apollo, PA
Babcock & Wilcox; Parks Township, PA
PP Chemicals America, Inc.; Lima, OH
Cabot Corporation; Boyerton, PA
Cabot Corporation; Reading, PA
Cabot Corporation; Revere, PA
Chemetron Corporation, Bert Avenue; Cleveland, OH
Chemetron Corporation, Harvard Avenue; Cleveland, OH
Chevron Corporation (formerly Gulf United Nuclear Fuels Corporation); Pawling, NY
Clevite; Cleveland, OH
Dow Chemical Company; Bay City and Midland, MI
Elkem Metals, Inc.; Marietta, OH
Engelhard Corporation; Plainville, MA
Fansteel, Inc.; Muskogee, OK
Hartley and Hartley (Kawkawlin) Landfill; Bay County, MI
Heritage Minerals; Lakehurst, NJ
Horizons, Inc.; Cleveland, OH
Kerr-McGee; Cimarron, OK
Kerr-McGee; Cushing, OK
Lake City Army Ammunition Plant (formerly Remington Arms Company);
Independence, MO
Magnesium Elektron; Flemington, NJ
Minnesota Mining and Manufacturing Co. (3M); Pine County, MN
Molycorp, Inc.; Washington, PA
Molycorp, Inc.; York, PA
Northeast Ohio Regional Sewer District/Southerly Plant; Cleveland, OH
Nuclear Metals, Inc.; Concord, MA
Old Vic, Inc.; Cleveland, OH
Permagrain Products; Media, PA
Pesses Company, METCOA Site; Pulaski, PA
Pratt and Whitney; Middletown, CT
RMI Titanium Company; Ashtabula, OH
RTI, Inc. (formerly Process Technology of North Jersey, Inc.; Rockaway, NJ
Safety Light Corporation; Bloomsburg, PA
Schott Glass Technologies; Duryea. PA
Sequoyah Fuel Company; Gore, OK
Shieldalloy Metallurgical Corporation; Cambridge, OH
Shieldalloy Metallurgical Corporation; Newfield, NJ

---

[1]The Allied Signal Aerospace, Bendix Division, in Teterboro, NJ, site and the Budd Company site in Philadelphia, PA, have been decommissioned and the sites have been removed from the SDMP list.

Contaminated List (continued)

Texas Instruments, Inc.; Attleboro, MA
UNC Recovery Systems; Wood River Junction, RI
Watertown Arsenal/Mall; Watertown, MA
Watertown GSA; Watertown, MA
Westinghouse Electric Corporation; Waltz Mill, PA
West Lake Landfill; St. Louis, MO
Whittaker Corporation; Greenville, PA
Wyman-Gordon Company; North Grafton, MA

ENFORCEMENT PROGRAM

Background:

As the federal agency responsible for regulating the civilian uses of nuclear materials, the NRC has an extensive program with many requirements. These requirements are imposed on 112 nuclear power plant licensees and approximately 8000 materials licensees. The requirements are stringent and technically demanding. Inevitably, with such an elaborate regulatory program, violations of requirements occur, through oversight, negligence, ignorance, confusion, and, in some instances, willful misconduct. The Commission has developed an enforcement program that seeks to promote and protect the public health and safety by ensuring compliance with the Atomic Energy Act, the Energy Reorganization Act, NRC regulations, and license conditions, obtaining prompt correction of violations and adverse quality conditions that may affect safety, deterring future violations, and encouraging improvement of licensee performance.

Program Operation:

The enforcement program starts with inspections and investigations to determine whether licensed activities are being conducted in compliance with regulatory requirements. All violations are subject to civil enforcement action. Following identification of a potential violation, it is assessed in accordance with the Commission's Enforcement Policy. This Policy has been approved by the Commission and is published as Appendix C to 10 CFR Part 2 of the Commission's regulations. As a policy and not a regulation, the Commission is able to depart from the Policy if circumstances warrant, but in practice, this happens only rarely. Violations that are done willfully are subject to criminal enforcement action. These cases are also investigated by the NRC's Office of Investigations (OI) and, if wrongdoing is found, the case is referred to the Department of Justice for consideration for prosecution.

There are three primary enforcement sanctions available: Notices of Violation, civil penalties, and orders. A Notice of Violation (NOV) summarizes the results of an inspection and formalizes a violation. It states the requirement and how that requirement was violated. A civil penalty is a monetary fine issued under authority of section 234 of the Atomic Energy Act. That section provides for penalties of up to $100,000 per violation per day. NOVs and civil penalties are issued based on violations. Orders may be issued for violations, or in the absence of a violation, because of a safety issue.

The Commission's order issuing authority is broad and extends to any area of licensed activity that affects the public health and safety. Orders modify, suspend, or revoke licenses. As a result of a recent rulemaking, the Commission can now issue orders to individuals who are not themselves licensed.

In addition to the primary sanctions, the NRC has administrative mechanisms available, including Confirmatory Action Letters to confirm in writing a licensee's commitment to take certain immediate corrective actions, and Notices of Deviation that address violations of non-legally binding commitments, such as an industry good practice or standard.

The first step in the enforcement process is assessing the severity level of the violation. Severity levels range from Severity Level I, for the most significant violations, to Severity Level V for those of minor concern. Eight supplements to the Enforcement Policy provide guidance in determining these severity levels. A higher severity level may be assigned for cases involving a group of violations with the same root cause, repetitive violations, or willful violations.

Enforcement conferences are held for violations assessed at Severity Levels I, II, or III, and may be held for violations assessed at Severity Level IV if increased management attention is warranted (e.g., repetitive violations). An enforcement conference is a meeting between the NRC and the licensee to (1) discuss the apparent violations, their significance, the reason for their occurrence, including the apparent root causes, and the licensee's corrective actions, (2) determine whether there were any aggravating or mitigating circumstances, and (3) obtain other information that will help the NRC determine the appropriate enforcement action. The decision to hold an enforcement conference does not mean that the agency has determined that a violation has occurred or that enforcement action will be taken. In accordance with the Enforcement Policy, enforcement conferences are normally closed to the public. However, on July 10, 1992, the Commission implemented a two-year trial program to allow certain enforcement conferences to be open to public observation. (See further discussion addressed under "Current Developments.")

Following the enforcement conference, the regional office prepares the proposed enforcement action. All Severity Level I and II cases, and some Severity Level III cases are sent to Headquarters for processing and approval. Routine Severity Level III materials cases, and Severity Level IV and V matters are issued directly from the regional office.

In the absence of mitigating circumstances, civil penalties are normally issued for Severity Level III or higher violations and may be issued for violations at Severity Level IV if the violations are repetitive or similar to previous Severity Level

IV violations. In addition, a civil penalty may be issued for any willful violation.

If a civil penalty is to be proposed, the base value of the penalty must first be determined. The base value is based on a combination of the type of licensed activity, the type of licensee, and the severity level of the violation. Once the base value is determined, a number of factors are considered that may either escalate or mitigate the amount of the civil penalty, depending on the unique circumstances of the case. These factors are: (1) who identified the violation, (2) was the corrective action prompt and extensive or untimely and only marginally acceptable, (3) how was the past performance of the licensee, (4) did the licensee have prior notice of similar events or other indications that should have alerted management, (5) were there multiple examples of the violation, and (6) what was the duration of the violation.

If a civil penalty is to be proposed, a written Notice of Violation and Proposed Imposition of Civil Penalty is issued and the licensee has 30 days to respond in writing, by either paying the penalty or contesting it. The NRC considers the response, and if the penalty is contested, may either mitigate the penalty or impose it by an order.

If the civil penalty is to be imposed by order, the order is published in the *Federal Register*. Thereafter, the licensee may pay the civil penalty or request a hearing.

In addition to civil penalties, orders may be used to modify, suspend, or revoke licenses. Orders that modify a license may require additional corrective actions, such as removing specified individuals from licensed activities or requiring additional controls or outside audits. The NRC issues a press release with a proposed civil penalty, order, or Confirmatory Action Letter.

Current developments:

On September 16, 1991, the Commission implemented a new rule that would allow orders to be issued to individuals. The deliberate misconduct rule applies to a licensee, an employee of a licensee, a contractor, subcontractor, or employee of them who knowingly provides to a licensee or contractor components or any other goods or services that relate to licensed activities. This rule prohibits (1) engaging in deliberate misconduct that causes, or but for detection would have caused, a licensee to be in violation of any NRC requirement, or (2) deliberately submitting to NRC, a licensee or contractor, or subcontractor, information known to be incomplete or inaccurate in some respect material to the NRC. Deliberate misconduct means an intentional act or omission that the person knows would cause or is a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee or contractor, whether or not the person knew a resulting violation of NRC requirements would occur.

An order issued under the deliberate misconduct rule might order the wrongdoer to remain out of licensed activities for a specified period, to notify the NRC before resuming involvement in licensed activities, or to inform any prospective employer of the issuance of the order. The order might require the employer to remove or confirm the removal of an employee from licensed activities, require the employer to advise prospective employers of the existence of the order when they call for reference checks, or require notice to the NRC if a licensee employs or desires to reemploy a wrongdoer in licensed activities.

Parties affected by orders may request a hearing before an Administrative Law Judge or a panel of the Atomic Safety and Licensing Board. Further appeal to the Commission, and ultimately the Court of Appeals, is possible.

In the recent rulemaking, the Commission also established a clearer mechanism for obtaining information from a licensee when the NRC is considering enforcement action. A Demand for Information may be issued to a licensee, requiring submission of a response to specific questions.

On July 10, 1992, the Commission implemented a two-year trial program to allow certain enforcement conferences to be open to public observation. Under the trial program, approximately 25 percent of all eligible enforcement conferences will be open to public observation, at least one conference will be open in each regional office, and open conferences will be conducted with a variety of types of licensees. Conferences that will be open to public observation will be selected on a random basis by selecting every fourth eligible conference involving a commercial operating reactor, hospital, or other type of licensee. In addition, conferences may also be open in cases where there is an ongoing adjudicatory proceeding with one or more intervenors.

Enforcement conferences will not be open to public observation if the enforcement being action being contemplated (1) may be taken against an individual or, if the action, though not taken against the individual turns on whether an individual has committed wrongdoing; (2) involves significant personnel failures where the NRC has requested that the individual(s) involved be present at the conference; (3) is based on the findings of an NRC OI report; or (4) involves safeguards, Privacy Act, or proprietary information. Enforcement conferences will also be closed if the conference will be conducted by telephone or at a relatively small licensee's facility.

The NRC intends to announce open enforcement conferences to the public normally 10 working days in advance of the conference through the following mechanisms: (1) notices posted in the Public Document Room; (2) toll-free telephone messages; and (3) toll-free electronic bulletin board messages. Pending establishment of the toll-free message systems, the public may call (800) 952-9674 to obtain a recording of upcoming open enforcement conferences.

Contact:

James Lieberman, Director, Office of Enforcement
US NRC, Washington, DC 20555. (301) 504-2741.

HIGHLIGHTS OF ENFORCEMENT PROGRAM

- Enforcement program seeks to protect public health and safety by ensuring compliance and correction of violations and deterring future violations.
- Violations are detected through inspections and investigations.
- Violations are subject to civil enforcement action and may be subject to criminal prosecution.
- Civil enforcement sanctions include: Notices of Violation, civil penalties, and orders.
- Severity level of a violation reflects the significance of the violation and ranges from the most significant, Severity Level I, to the least, Severity Level V.
- Civil penalties are normally issued for Severity Level III or higher violations.
- The amount of a civil penalty assessed varies with type of licensed activity, type of licensee, severity level, and escalation and mitigation factors.
- If a civil penalty is proposed, licensee may respond by paying or contesting the action.
- If licensee protests action, staff considers response, and either mitigates the penalty or imposes it by order.
- Licensee must then pay or request an administrative hearing.
- Orders may be used to modify, suspend, or revoke a license.
- Orders may also address deliberate wrongdoing by individual employees of licensees, contractors, or others who provide goods or services that relate to licensed activities.
- An order to an individual might remove him or her from licensed activity, require NRC notification of the individual's reemployment in licensed activities, or require notification to prospective employers of the existence of an order.
- NRC may use Demands for Information to obtain information when considering enforcement action.
- In a two-year trial program starting July 10, 1992, one of every four eligible enforcement conferences involving a commercial operating reactor, hospital, or other type of licensee will be open to public observation.
- A schedule of open enforcement conferences can be obtained by calling (800) 952-9674.

INSPECTION OF NUCLEAR POWER PLANTS

Background:

The primary safety consideration in the operation of any nuclear reactor is the control and containment of radioactive material, under both normal and accident conditions. Numerous controls and barriers are installed in reactor plants to protect workers and the public from the effects of radiation.

Both the industry and the NRC have roles in providing these protections and in ensuring that they are maintained. The NRC establishes regulations and guides for the construction and operation of nuclear reactors. Organizations licensed by the NRC must abide by these regulations and are directly responsible for designing, constructing, testing, and operating their facilities in a safe manner. The NRC, through its licensing and inspection programs, provides assurance that its licensees are meeting their responsibilities.

Inspection Program:

The responsibility for safe operation of a nuclear power plant lies with the licensee. The NRC inspection program is designed, through selective examinations, to ensure that the licensee is meeting his responsibility. The NRC inspection program is audit-oriented to verify, through scrutiny of carefully selected samples, that relevant activities are being properly conducted and equipment properly maintained so as to ensure safe operations. What to sample, sample size, and the frequencies of inspection are all judgments based on the importance of the activity or system to overall safety and on available resources. The inspection program is preventive in nature and is intended to anticipate and preclude significant events and problems by identifying underlying safety problems. The inspection process monitors the licensee's activity and gives feedback to the licensee's management for appropriate corrective action. However, implementation of the NRC inspection program does not supplant the licensee's programs or attenuate its responsibilities. What the inspection program seeks to provide is a feedback mechanism and an independent verification of the effectiveness of the licensee's implementation of its programs, to ensure that operations are being carried out safely and in accordance with applicable NRC requirements. Inspections are performed on power reactors under construction, in test conditions, and in operation. The inspections are conducted primarily by region-based and resident inspectors. Resident inspectors are stationed at each reactor under construction and in operation. Region-based inspectors operate out of the five Regional Offices located in or near Philadelphia, Atlanta, Chicago, Dallas, and San

Francisco. These programs are supplemented by inspections conducted by special teams made up of personnel from both NRC Headquarters and Regional Offices.

Inspections are a vital part of NRC's review of applications for licenses, and also of the process leading to issuance of construction permits and operating licenses. Inspections continue throughout the operating life of a nuclear facility.

Prior to construction, the inspection program concentrates on the applicant's establishment and implementation of a quality assurance program. Inspections cover quality assurance activity related to design, procurement and planning for fabrication and construction of the facility.

During construction, samples taken across the spectrum of licensee activity are examined to confirm that the requirements of the construction permit issued by the NRC are being followed and that the plant is being built according to the approved design and applicable codes and standards. Construction inspectors look for qualified personnel, quality material, conformance to approved design, and a well-formulated and -implemented quality assurance program. As construction nears completion, pre-operational testing begins, in order to demonstrate the operational readiness of the plant and its staff. Inspections during this phase seek to determine whether the licensee has developed adequate test plans - both to ensure that tests are consistent with NRC requirements, and to ascertain whether the plant and its staff are thoroughly prepared for safe operation. Inspections during the pre-operational phase involve (1) reviewing overall test procedures; (2) examining selected test procedures for technical adequacy; and (3) witnessing and assessing selected tests to verify that test objectives have been met and to confirm the consistency of planned and actual tests. Inspectors also review the qualifications of operating personnel and verify that operating procedures and quality assurance plans are properly developed and implemented.

About six months before the operating license is issued, a startup phase begins, preparatory to fuel loading and "power ascension." After issuance of the operating license, fuel is loaded into the reactor and the actual startup test program begins. As in pre-operational testing, NRC inspection emphasis is given to test procedures and results. The licensee's management system for startup testing is appraised, test procedures are analyzed, tests are witnessed, and licensee evaluations of test results are reviewed. Thereafter, the NRC continues its inspection program for the rest of the operating life of the plant.

The staff is developing a new construction inspection program for reactors to be built under combined construction and operating licenses issued under 10 CFR Part 52. The new inspection program will continue to verify the safety aspects of a plant's construction and testing, as described above for the current program, and will allow for more systematic inspection planning and documentation of inspection results. Also, the new construction inspection program will be structured to ensure verification of satisfactory completion by licensees of the Inspections, Tests, Analyses and Acceptance Criteria (ITAACs) included in a combined license and required by 10 CFR Part 52.

The NRC assures that the licensee is operating safely through selective inspections. An on-site resident inspector provides a continual inspection and regulatory presence, as well as a direct contact between NRC management and the licensee. The activity of the resident inspector is supplemented by the work of engineers and specialists from the Regional Office staff who perform inspections in a wide variety of engineering and scientific disciplines, ranging from civil and structural engineering to health physics and reactor core physics.

The inspection program for operating reactor plants is defined in the NRC Inspection Manual, in terms of its frequency, scope and depth. Detailed inspection procedures provide instructions and guidance for NRC inspectors. The program consists of three major elements: core inspections - the minimum required at all plants; "area of emphasis" inspections - special inspections to focus on a specific issue; and regional initiative inspections - those required to resolve safety issues brought to light by other inspections or as a result of plant operational experience. The program is structured to ensure that, among other considerations, the resources available for inspection are used efficiently and effectively, with particular attention accorded those plants where, based on past performance, improvements in the levels of protection and safety-consciousness may be in order.

The inspection program is designed to ensure that nuclear power plants are constructed and operated safely and in compliance with regulatory requirements. Its results are factored into NRC's overall evaluation of licensee performance under the Systematic Assessment of Licensee Performance (SALP) program. When a safety problem or failure to comply with requirements is discovered, the NRC requires prompt corrective action by the licensee. Appropriate enforcement action is taken in accordance with the NRC's enforcement policy.

The NRC conducts periodic self-assessments of the inspection program to evaluate its effectiveness in achieving its regulatory objectives.

In February 1989, the Commission announced a policy of cooperation with States which allows States to observe, and in some cases participate in, NRC inspections at reactor facilities.

Contact:

Mark W. Peranich, Chief, Operating Reactor Inspection Section
Inspection and Licensing Policy Branch
US NRC, Washington, DC 20555 (301) 504-3078

HIGHLIGHTS OF INSPECTION PROGRAM

- NRC, through its licensing and inspection programs, ensures licensees are meeting their responsibilities for constructing and operating nuclear reactors in a safe manner.

- NRC inspection program is independent, audit oriented, and preventive in nature to anticipate and preclude significant events by identifying underlying issues.

- Inspection process provides feedback to the licensee's plant management to allow it to take appropriate corrective action.

- Inspections performed by resident, regional, and headquarters inspectors.

- Inspections performed prior to construction, during construction, during preoperational testing, and routinely after the plant is in operation.

- Inspection program defined in NRC Inspection Manual and detailed inspections procedures exist for:

    1) core inspections - the minimum done at all plants;

    2) area of emphasis inspections - special inspections to focus on a specific issue;

    3) initiative inspections - those which are required to adequately resolve safety issues.

- Inspection results factored into the overall evaluation of licensee performance under the Systematic Assessment of Licensee Performance (SALP) program.

- NRC conducts self-assessments of the inspection program.

ENVIRONMENTAL IMPACTS OF NUCLEAR POWER PLANTS

Background:

The discharge of radioactive effluents from the routine operation of nuclear power plants can result in environmental impacts. These impacts can be on man, and terrestrial and aquatic biota. In nearly all cases a Final Environmental Statement (FES) was issued by the NRC which details the potential impacts resulting from the routine operation of each nuclear power plant. The NRC regulation, 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," requires that each license authorizing operation of a nuclear power reactor include technical specifications that (1) keep releases of radioactive material to unrestricted areas during normal operation as low as is reasonably achievable, and (2) comply with the applicable provisions of 10 CFR 20.106, "Radioactivity in Effluents in Unrestricted Areas" or Subpart D, "Radiation Dose Limits for Individual Members of the Public," of the revised Part 20. These and other NRC regulations require NRC licensees to have in place various effluent and environmental monitoring programs to ensure that the impacts are minimized.

Environmental Impacts:

Radioactive effluent releases and their associated doses are reported by licensees in Semiannual Radioactive Effluent Release Reports (recently changed to an annual report) while radioactivity levels in various environmental media are reported in the Annual Environmental Operating Reports (AEOR). The Radioactive Effluents Release Reports include the amount of liquid and airborne radioactive effluents discharged and the calculated doses for the period of release. Doses which are typically presented for airborne effluents include the beta and gamma air doses from noble gases and the maximum organ dose from radioiodine and particulates. These doses are compared to the design objective doses of Appendix I to 10 CFR 50. For liquid effluents, the total body and maximum organ doses are typically presented and are also compared to the design objective doses of Appendix I.

The AEOR provides the results of an environmental sampling and analysis program which is focused on the radiation exposure pathways specific to the given plant. Typical sampling programs include an inner and outer ring of TLDs (thermoluminescent dosimeters) around the plant; airborne radioiodine and particulate samplers; samples of surface, groundwater, and drinking water and downstream shoreline sediment from existing or potential recreational facilities; and samples of ingestion pathway sources such as milk, fish and invertebrates, and food products including broad leaf vegetation. The results of the AEOR report are used to supplement the effluent monitoring program to ensure that potential impacts do not go undetected.

Regulations:

The regulations, which are presently in place to limit offsite releases and their associated radiation doses, are much more restrictive than those initially issued and those which the first licensed nuclear power plants (during the period of the 1960s) were required to meet.

On May 5, 1975, the NRC amended regulations 50.34a and 50.36a and added a new Appendix I to Part 50 which provided numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as is reasonably achievable." The adoption of these regulations meant that the limiting criterion for nuclear power plant effluents was no longer Part 20, but the design objectives of Appendix I.

An additional regulatory requirement was placed on uranium fuel cycle licensees with the provisions of 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," which was promulgated by the Environmental Protection Agency on December 1, 1979. These standards established total body, thyroid, and other organ dose limits for all effluents and direct radiation except radon and its daughters. The NRC subsequently incorporated the EPA regulations into 10 CFR Part 20 by reference on March 25, 1981.

Current Status:

In addition to the Annual Effluent Release Reports and the AEOR, the NRC uses other means of verifying that licensees fully evaluate the potential impacts of their operations. The NRC contracts with some 35 states which perform various environmental monitoring programs including environmental sampling and analysis around nuclear power plants. The NRC also has a mobile laboratory which is used during plant inspections to confirm, using split samples, the accuracy of the licensee's radiological effluent monitoring program. Licensees are also required to participate in an Interlaboratory Comparison Program which provides an independent check of the accuracy and the precision of the measurement of radioactive material in environmental samples. The NRC also conducts periodical aerial surveillance of nuclear power plants in which measurements of direct radiation are made.

The NRC documents the results of its independent monitoring and assessment efforts in plant specific inspection reports and in reports such as NUREG/CR-2850, "Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites," and

NUREG-0837, "NRC TLD Direct Radiation Monitoring Network." The NRC also compiles licensees' effluent reports and publishes the results in NUREG/CR-2907, "Radioactive Materials Released from Nuclear Power Plants."

With the implementation of the design objective doses of Appendix I and their associated technical specifications, the effluents released from nuclear power plants have decreased significantly. The collective annual radiation dose to the public has decreased from 1300 person-rem in 1975 (the initial year for which such evaluations were published) to 84 person-rem in 1989. When normalized to the electric power generation, collective doses have decreased from 7.6 to 0.16 person-rem/TW-hr (10^{12} watts). Collective dose represents the sum of doses to individuals within 80 km (50 mi) of each facility. A significant contributor to the reduction in airborne effluents has been the addition of the Augmented Offgas Systems to boiling water reactors. Reductions have also resulted from improved fuel performance and licensees' improved effluent control programs.

Contact:
LeMoine J. Cunningham, Chief, Radiation Protection Branch
U.S. NRC, Washington, DC 20555 (301) 504-1086

ENVIRONMENTAL IMPACTS OF NUCLEAR POWER PLANTS

- Final Environmental Statements on each plant detail the potential impacts resulting from routine operation.
- Licensees report radiation releases, including liquid and airborne, and their associated doses in the Semiannual (recently changed to Annual Effluent Release Report).
- Licensees report radioactivity levels from the environmental sampling of air, ground, water, and ingestion pathways in the Annual Environmental Operating Reports.
- 10 CFR 50.36a requires each license for a nuclear power reactor include technical specifications that not only keep releases of radioactive material to unrestricted areas as low as reasonably achievable, but also comply with 10 CFR 20, Subpart D (old 10 CFR 20.106).
- 10 CFR 20, Subpart D establishes dose limits due to radioactive materials in effluents in unrestricted areas.
- The NRC documents licensee's effluent releases and the results of the staff's independent monitoring and assessment effort in NUREG-2907, NUREG-2850, NUREG-0837, and plant specific inspection reports.
- 40 CFR 190 sets standards for annual doses to a member of the public for uranium fuel cycle licensees.
- The staff verifies that licensees evaluate potential radiological impacts through onsite inspections, a mobile lab, state contracts, interlaboratory comparison, and aerial surveillance.
- 10 CFR 50.34a & Appendix I provide numerical guides for design objectives and limiting conditions for operation to meet as low as is reasonably achievable criteria.
- Effluent releases result in very small doses to members of the public living around nuclear power plants.

URANIUM MILL TAILINGS

Background:

A lack of orders for new nuclear power plants and the importation of uranium from other countries have now severely eroded the value of uranium and resulted in most U.S. uranium mills shutting down operations or operating on a limited basis.

Many mills are, or will be, conducting reclamation of tailings piles created in the process of extracting source material (in the form of "yellow cake") from uranium-bearing ore. These mill tailings wastes, both from unlicensed inactive mills (formerly used in providing uranium primarily for the weapons program) and from licensed active mills regulated by the Nuclear Regulatory Commission or the Agreement States, pose a long-term hazard to the public health and safety. To provide for the disposal, long-term stabilization, and control of these uranium mill tailings in a safe and environmentally sound manner, Congress enacted the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA).

In terms of health hazard, the most hazardous radioactive constituent in uranium mill tailings is radium, which has a very long half-life. Radium, besides being hazardous itself, produces radon, a radioactive gas whose decay products can cause lung cancer. This makes mill tailings hazardous for thousands of years.

UMTRCA established two programs to protect public health and the environment from uranium mill tailings. The Title I program established a joint Federal/State-funded program for remedial action at abandoned mill tailings sites, with ultimate Federal ownership under license from NRC. Under Title I, NRC must evaluate the Department of Energy's (DOE) designs and concur that DOE's actions for cleanup and remediation of these inactive tailings sites meet standards set by the Environmental Protection Agency (EPA). The Title II program is directed towards the active mill tailings sites (those sites under license by NRC or Agreement States). Title II provides: (1) NRC authority to control radiological and nonradiological hazards; (2) EPA authority to set generally applicable standards for both radiological and nonradiological hazards; and (3) eventual State or Federal ownership, under license from NRC. For Agreement States, NRC also is required to make a determination that all applicable standards and requirements have been met by uranium mills licensed by Agreement States before termination of their licenses.

Regulations and Standards:

UMTRCA charged the EPA with the responsibility for promulgating standards for control of uranium mill tailings. In 1983, EPA issued standards in 40 CFR Part 192 for both Title I sites (Subparts A, B, and C) and Title II sites (Subparts D and E). The portion of the EPA Title I standards dealing with ground water requirements were remanded by the Tenth Circuit Court of Appeals on September 3, 1985. Based on this court decision, EPA was directed to promulgate new groundwater standards. EPA proposed these standards in the form of revisions to Subparts A-C of 40 CFR Part 192 in September, 1987, and now is in the process of completing action to promulgate the final groundwater standards.

For Title II sites, NRC conformed its standards in 10 CFR Part 40 to be consistent with the applicable EPA standards. NRC's final regulations conforming to EPA's requirements for radiological and nonradiological protection and long-term stabilization of the impoundments for the tailings were published on October 16, 1985. NRC's final regulations addressing EPA's groundwater protection standards were published on November 13, 1987.

EPA, in developing its mill tailings standards, estimated that its standards would significantly reduce radon emissions from tailings and approximately 600 lung cancer deaths per century would be avoided. Since the EPA standards require that the impoundments for the tailings must be designed to be stable for 1,000 years, to the extent practicable, but in no case less than 200 years, it is assumed that the actual engineered structures will degrade slowly over possibly thousands of years. Therefore, the use of the standards and NRC's implementing regulations could result in the avoidance of tens of thousands of calculated radon-related lung cancer deaths.

Current Status:

Title I -- Reclamation Work at Inactive Tailings Sites

Twenty-four inactive mill tailings piles at 22 sites designated by DOE range in size from about 60 thousand to 4.6 million cubic yards of material. Except for a site at Canonsburg, PA, the inactive sites are located in the western United States (Arizona, Colorado, Idaho, New Mexico, North Dakota, Oregon, Texas, Utah, and Wyoming). The DOE surface remediation program is estimated to cost approximately $1.3 billion and is expected to be completed by 1998. The DOE groundwater cleanup phase was initiated by DOE in 1991. DOE has completed an internal draft of the Programmatic Environmental Impact Statement (PEIS) for its groundwater remediation phase. The NRC has reviewed and commented on this draft, as part of its cooperating agency status in the PEIS process. The final PEIS should be released for public comment in early 1994. DOE's initial cost estimates for groundwater remediation were over $500 million with a completion date of 2020. DOE has completed remedial work, with the exception of groundwater cleanup, at approximately half of the Title I sites. The DOE became a licensee of the NRC under 10 CFR 40.27 with the completion and NRC concurrence of the Long-Term Surveillance Plan for the Spook, Wyoming inactive tailings site. It is anticipated that in 1994, 3 to 5 more completed mill tailings sites will come under the general license in 40.27 for the long-term custodial care by DOE.

Title II -- Licensed Mill Tailings Sites

Of 27 NRC licensed uranium recovery facilities, there are 19 mills; 5 in situ leach facilities; 1 mine-backfill; 1 ion-exchange facility and 1 heap leach. None of the uranium mills are operating. Three of the in situ leach facilities are presently operating. There are four mills and two in situ facilities that have standby licenses which would permit these facilities to go into operation in a relatively short time. Most of the NRC licensed facilities are either expected to begin, or have already started, reclamation activities to provide long-term stabilization and closure of the tailings impoundments. While NRC has terminated in situ leach facility licenses, it has not, as yet, terminated a license for any uranium mill facility. These NRC-licensed sites are located in Nebraska, New Mexico, Utah, and Wyoming. There also are 10 uranium mills in Agreement States (Colorado, Texas, and Washington) that have similar non-operational tailings impoundments.

In the fall of 1991, NRC, EPA, and the affected mill tailings Agreement States agreed that there was a need to eliminate the dual regulation created by NRC's authority under UMTRCA and the Atomic Energy Act of 1954, as amended, and EPA's authority under the Clean Air Act (CAA). This interagency consultation resulted in the execution of a Memorandum of Understanding (MOU) to provide the basis for eliminating the dual regulation by EPA under the CAA. Current activities are addressing the disposal standards in Subpart T of 40 CFR Part 61. The radon flux standards in Subpart T are the same as those under UMTRCA. Consequently, the primary focus of the MOU is to ensure that non-operational piles are closed to comply with the radon standards as expeditiously as practicable, with a goal that such closure occur by the end of 1997. The MOU specifies that the schedules for closure be enforceable by NRC or the affected Agreement States. The MOU further provides that the dual regulation of operational sites under Subpart W of Part 61 will be addressed subsequently.

In the late summer of 1992, NRC was requested by EPA, the American Mining Congress, the Natural Resources Defense Council, and the Environmental Defense Fund to sign a settlement agreement related to EPA's Part 61, Subpart T standards and associated stays of the standards. Although NRC is not a party in the underlying lawsuits, the Commission recognized that many of the actions outlined in the settlement agreement requiring NRC or Agreement State action were an integral part of the agreement. While the Commission did not support signing the settlement agreement, it did direct staff to prepare a letter to the affected parties in which the Commission would indicate it was in general accord with the substantive provisions of the draft final settlement agreement dated November 24, 1992.

A major milestone identified in the MOU, was approval by September 30, 1993, of all detailed licensee reclamation plans to construct final radon barriers. On October 18, 1993, we wrote to EPA that NRC and Agreements States had approved all the plans except that for the Atlas mill site at Moab, Utah whose reclamation plan was the subject of extensive public comment.

<u>Contact</u>:

John J. Surmeier, Chief, Uranium Recovery Branch
U.S. NRC, Washington, DC 20555
(301) 504-3439

URANIUM MILL TAILINGS HIGHLIGHTS

- Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA) established a comprehensive regulatory framework for all uranium mill tailings for long-term custody and control.

- Title I of UMTRCA established a joint Federal-State funded program for remedial action at sites designated by Congress as "inactive" as of 1978, with ultimate Federal ownership under license from NRC.

- The 24 Title I tailings piles, except for the site at Canonsburg, PA, are located in the western United States. The DOE surface remediation work is estimated to cost approximately $1.3 billion and to be completed by 1996. DOE anticipates that all of these sites will be under license from NRC by 1998. The DOE groundwater cleanup phase was initiated in 1991. DOE has completed a draft of a Programmatic Environmental Impact Statement (PEIS) for this groundwater cleanup phase, which has been reviewed by NRC. The final PEIS should be released for public comment in early 1994. Groundwater cleanup has been estimated by DOE to cost over $500 million.

- Title II of UMTRCA provided: (1) NRC authority to control radiological and nonradiologial hazards; (2) EPA authority to set generally applicable standards for both radiological and nonradiological hazards; and (3) eventual State or Federal ownership under license from NRC.

- There are 27 uranium recovery facilities licensed by NRC under 10 CFR Part 40 in conformance with EPA generally applicable standards in 40 CFR Part 192.

- There are 15 NRC licensed and 8 Agreement State-licensed facilities that have non-operating mill tailings impoundments.

- An NRC-EPA Agreement State MOU of October 1991 provides the basis for eliminating dual regulation by EPA under the Clean Air Act and establishes a schedule for closure of the tailings impoundments at these 23 non-operating sites by the end of 1997.

STATE COMPLIANCE WITH 1993 MILESTONE AND 1996 LEGISLATIVE OBJECTIVE OF THE LOW-LEVEL RADIOACTIVE WASTE POLICY AMENDMENTS ACT OF 1985

Background:

The Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA) (Pub. L. 99-240) establishes a series of milestones, penalties, and incentives for regional compacts and States to promote progress toward being able to manage their low-level radioactive waste (LLW) by 1993. However, slow State progress in developing new LLW disposal facilities has resulted in the storage of LLW at some generator sites, even before January 1, 1993. This paper includes background information on the LLRWPAA and the status of LLW disposal facility development. Related regulatory actions are summarized, including a proposed rulemaking that would establish criteria for on-site storage of LLW after January 1, 1996.

LLW Disposal:

Low-level radioactive waste is a general term for a variety of radioactively contaminated wastes generated by nuclear power plants and related industries, hospitals, medical and educational research institutions, private and governmental laboratories, and other commercial activities that use radioactive materials as a part of their normal operations. Approximately 48,000 cubic meters (1.7 million cubic feet) of LLW were disposed of in 1992. LLW is currently disposed of using shallow land burial at privately operated facilities located in the States of South Carolina and Washington.

The Low-Level Radioactive Waste Policy Act of 1980 (LLRWPA) made the States responsible for the disposal of commercially generated and certain Federally generated LLW. The LLRWPA encouraged the States to form compacts to dispose of LLW on a regional basis. The LLRWPA also designated January 1, 1986, as the date after which compacts could restrict the use of their disposal facilities by excluding waste generated outside the compact region. However, by 1983, it had become clear that no new disposal facilities would be operational by the 1986 milestone. As a result, in January 1986, the LLRWPAA was enacted. The LLRWPAA extended the January 1, 1986, deadline by seven years, to January 1, 1993. By that date, new LLW disposal facilities were expected to be operational, and the rights of the LLW generators, to dispose of their LLW at the three sites then operating, would end.

At present, nine compacts have been formed, representing 42 States. The accompanying figure shows the current arrangements of compacts and unaffiliated States (i.e., those States not in a compact). Legislation to establish the Texas Low-Level Radioactive Waste Disposal Compact (Texas Compact) was signed by Texas Governor Ann Richards on June 9, 1993. The legislation provides that if either of the other two designated compact member States (Maine and Vermont) ratifies the compact, the agreement will be in full force for Texas and the ratifying State. Legislation ratifying the compact was enacted in Maine on June 21, 1993, and approved by a majority of Maine voters in a referendum held on November 2, 1993. The Vermont Legislature is expected to consider the compact legislation in January 1994. The U.S. Congress must then consent to the Texas Compact.

The LLRWPAA required the sited States of Nevada, South Carolina, and Washington to make disposal capacity available to LLW generators until December 31, 1992, subject to certain conditions. Although the Washington facility remains open, serving the Northwest Compact (its compact) and, under contract, the Rocky Mountain Compact, the Nevada facility stopped accepting LLW on December 31, 1992. The facility in South Carolina will remain open until January 1, 1996, subject to various conditions. However, the facility will close permanently to out-of-region waste on July 1, 1994. Widespread storage is expected after this date.

The importation of out-of-region waste to the South Carolina facility, during the period of January 1, 1993, to June 30, 1994, will not be approved for States or compact regions that are not making adequate progress toward providing for disposal of their own LLW. The States of Michigan, New Hampshire, and Rhode Island, and the Commonwealth of Puerto Rico are not eligible for access to the South Carolina facility under these conditions. Michigan generators began storing LLW in November 1990, when they were denied access because Michigan was no longer in compliance with the LLRWPAA. New Hampshire's Seabrook Nuclear Power Plant has also been storing LLW since October 1990, because a contract with the Rocky Mountain compact excluded it.

To help ensure that the States make adequate progress to develop new LLW disposal facilities, the LLRWPAA established six milestones by which the States should make decisions and commit to certain actions. The majority of the States met the requirements of the three milestone dates that had passed by January 1990. However, only the Central, Central-Midwest, and Southwestern Compacts met the January 1, 1992, milestone requirement, where their respective Host States of Nebraska, Illinois, and California submitted applications for disposal facilities. The State of Texas conformed to this milestone on March 2, 1992, by submitting a disposal facility license application. However, no State had a new LLW disposal facility operational on January 1, 1993, as envisioned by the LLRWPAA. The remaining milestone of the LLRWPAA required that on January 1, 1996, States without access to operating disposal sites, upon proper notice by the generator or owner, take title to and possession of LLW. However, this section of the LLRWPAA, often referred to as the "take-title" provision, was held to be unconstitutional by the U.S. Supreme Court, on June 19, 1992.

Progress to Develop New LLW Disposal Facilities:

Only two new facilities are now scheduled to be operational by January 1996, those in California and North Carolina, and the latter will replace the existing Barnwell facility. LLW disposal facilities in the Host States of Nebraska, New Jersey, and Texas are forecast to be operational between the period 1996 and 1998. Six other sites are scheduled to begin operation during the period 1999 to 2001. The unaffiliated States of Michigan, New Hampshire, and Rhode Island, as well as the District of Columbia and the Commonwealth of Puerto Rico do not have a disposal site under development. A number of States believe that they may be able to fulfill their responsibilities through the contracting and/or compact process. The accompanying table shows the dates by which compact Host States and unaffiliated States accomplished, or expect to accomplish, key steps in developing new disposal facilities.

Recent events may significantly affect the schedule for the operation of the California low-level radioactive waste disposal site. In a November 24, 1993, letter to California Governor Pete Wilson, Secretary of the Interior Bruce Babbitt declared he would postpone any further action on the transfer of federal lands for the proposed Ward Valley site in California. Secretary Babbitt cited a challenge in California state court concerning the issuance of the license for the construction and operation of the Ward Valley facility. In August 1993, Secretary Babbitt had stated he was prepared to transfer the Ward Valley site after a hearing of several months duration was conducted. Governor Wilson provided Interior Secretary Babbitt with a detailed proposal for the requested hearing in September.

On November 18, 1993, Senator Boxer of California announced the formation of the W.A.R.D. Task Force (Warning About Radioactive Danger), a group established to oppose the Ward Valley site. Senator Boxer also wrote to President Clinton, on November 20, opposing the proposed land transfer to the State by the Department of Interior. On December 8, Senator Boxer released a report on the Ward Valley site, prepared by three geologists who are employees of the U.S. Geological Survey (USGS), apparently acting as private citizens, citing concerns about the Ward Valley site.

Since no new LLW disposal facilities were operational by January 1, 1993, and the compact commissions that control the existing LLW disposal sites have either closed their facilities or set conditions on receiving LLW from outside their regional compacts, some licensees who generate LLW have been forced to store their LLW on-site, until disposal capacity is available. Nearly all the Governors' Certifications submitted to meet the 1990 milestone of the LLRWPAA indicated that the State planned on interim storage by waste generators during the 1993 through 1996 period. However, since the South Carolina facility is to be available to many generators until July 1, 1994, there will be a mixed pattern of disposal and storage during this time period.

Regulatory Actions:

Because of some States' and compacts' slow progress in meeting the January 1, 1996, milestone of the LLRWPAA, the Commission is concerned about the likelihood of indefinite widespread on-site storage of LLW. Although public health and safety can be adequately protected if LLW is stored, it will be enhanced by disposal, rather than long-term, indefinite, storage of LLW. Disposal of LLW in a limited number of facilities licensed under existing regulations (10 CFR Part 61 or compatible Agreement State regulations) will provide better protection of public health and safety and the environment than storage at hundreds of sites around the country. Permanent disposal of LLW has always been the preferred option for managing LLW, as reflected in the LLRWPAA. Because of these concerns the Commission has proposed to amend its regulations, to establish license condition requirements for on-site storage of LLW, by licensees, after January 1, 1996.

In this proposed rulemaking, the Commission has restated and emphasized its position that it will not look favorably upon on-site storage of LLW by generators after January 1, 1996. Under the proposed amendments, on-site storage of LLW would not be permitted after January 1, 1996 (other than reasonable short-term storage necessary for decay or for collection or consolidation for shipment off-site), unless the licensee documents that it has exhausted other reasonable waste management options. Such options include taking all reasonable steps to contract, either directly or through the State, for disposal of the waste.

This proposed rulemaking would supplement, but not supersede, the existing regulatory framework applicable to storage of LLW, and the conditions in themselves would not authorize on-site storage. On-site storage of LLW at reactors would continue to be subject to 10 CFR 50.59 evaluations (which allow licensees to store, provided there is no outstanding safety issue), as well as all other regulatory requirements currently in place. Licensees should continue to use appropriate regulatory guidance for on-site storage of LLW.

Current Status:

A proposed rulemaking was published in the Federal Register on February 2, 1993. The public comment period expired on April 5, 1993. The staff is currently summarizing and analyzing the comments received. Final action on this rulemaking is awaiting Commission action.

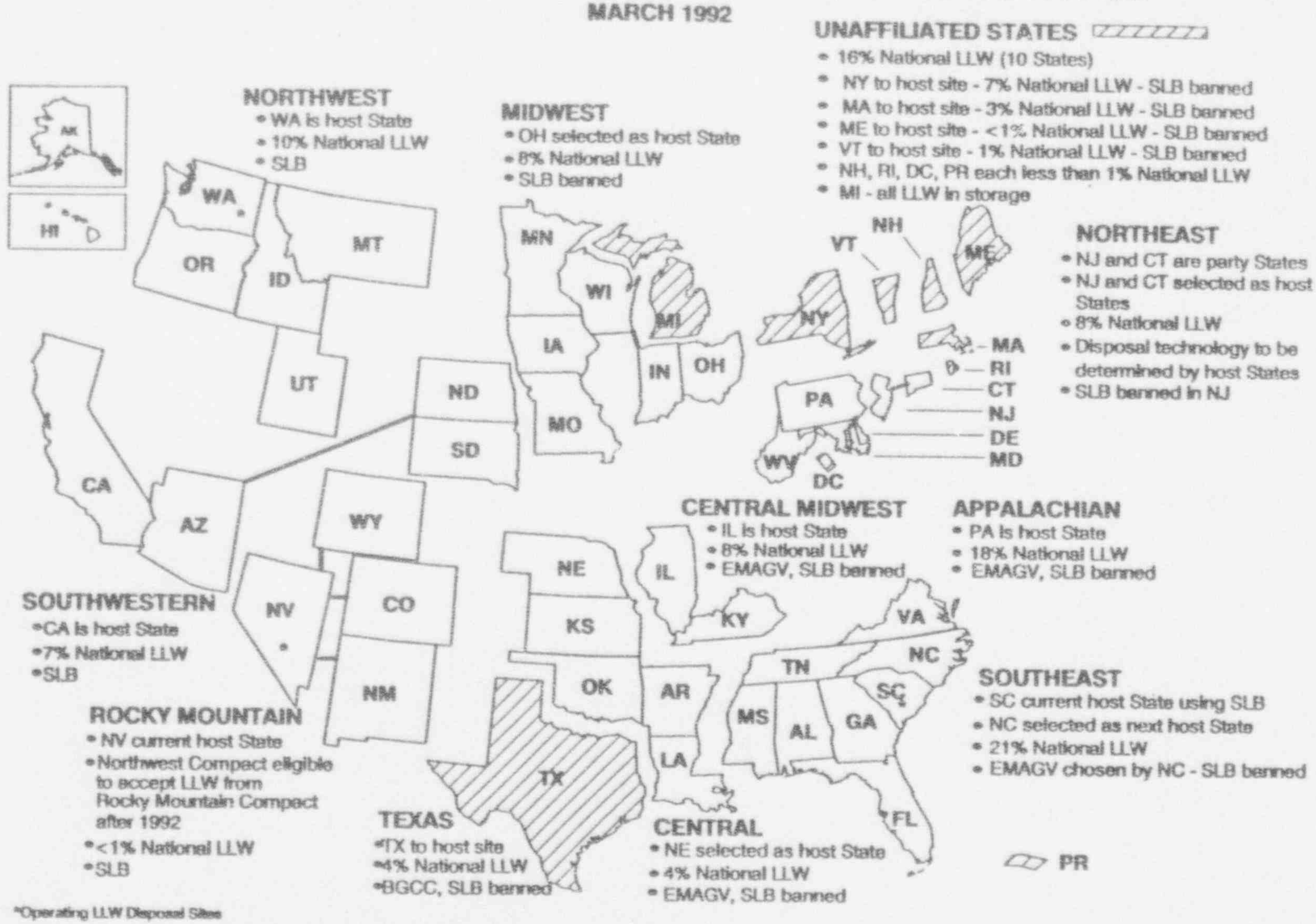
Highlights of this media briefing background paper can be found in the attachment.

Contact:

Michael Bell, Chief, Low-Level Waste Management Branch, Division of Low-Level Waste Management and Decommissioning, Office of Nuclear Material Safety and Safeguards, U.S. NRC, Washington, DC 20555 (301) 504-2553.

LOW-LEVEL RADIOACTIVE WASTE COMPACT STATUS

MARCH 1992



Note: National LLW volume for 1991 = 1.4 million cube feet disposed
SLB = shallow land burial
EMAGV = Earth mounded above grade vault
BGCC = below ground concrete canisters

Source: Office of State Programs, NRC

Figure 1

Actual and Estimated Dates for
Completing Steps in Facility Development
(Estimated Dates Obtained from Compacts/States)

Compact/Host State	Select Site	Submit License Application	Operate Facility
Appalachian/Pennsylvania	Early 1997	Early 1997	Mid-1999
Central/Nebraska	Dec 1989	Jul 1990	Sept 1998
Central Midwest/Illinois	Unscheduled	Oct 1997	June 2000
Midwest/Ohio	Unscheduled	Unscheduled	Unscheduled
Northeast/Connecticut &	Unscheduled	July 1997	Dec 1999
New Jersey	1994	Mid-1996	Late 1998
Southeast/North Carolina	End 1993	Dec 1993	Jan 1996
Southwest/California	Mar 1988	Dec 1989	late 1994
Unaffiliated States			
Maine (See note.)	1993	Unscheduled	Unscheduled
Massachusetts	Jun 1994	early 1998	2000
New York	1995/1996	1999	2001
Texas	Aug 1991	Mar 1992	1996
Vermont (See note.)	Feb 1994	Unscheduled	Unscheduled

Note: It is anticipated that Maine, Texas and Vermont will form a compact with Texas as the host state and will not remain unaffiliated (see text).

HIGHLIGHTS

STATE COMPLIANCE WITH 1993 MILESTONE AND 1996 LEGISLATIVE OBJECTIVE OF THE LOW-LEVEL RADIOACTIVE WASTE POLICY AMENDMENTS ACT OF 1985

- LLRWPAA established milestones, incentives, and penalties for States to develop new low-level radioactive waste (LLW) disposal facilities.
  - Milestones were established in 1986, 1988, 1990, 1992, 1993, and 1996.
  - Waste disposal surcharges and take-title and possession provisions were penalties for failure to comply.
  - Partial rebate of surcharges to States provided incentive in the form of financial assistance.
  - U.S. Supreme Court has held the 1996 take-title provision to be unconstitutional.
- Majority of States met the first three milestones.
- Only four States (California, Illinois, Nebraska, and Texas) met the 1992 milestone, and only California and North Carolina are scheduled to meet the 1996 legislative objective of the LLRWPAA.
- On-site storage of LLW at some generator sites is occurring because of lack of access to disposal facilities, especially in the States of Michigan (since November 1990), New Hampshire (Seabrook Nuclear Power Plant since October 1990), and Rhode Island, and the District of Columbia and Puerto Rico; and more recently the States of the Central Compact.
- Existing U.S. Nuclear Regulatory Commission guidance, in conjunction with current regulations, provides the regulatory and licensing framework for LLW storage.
- Although public health and safety can be protected if LLW is stored, public health and safety will be enhanced by disposal.
- NRC does not look favorably upon on-site storage of LLW.
- Commission has published a proposed rulemaking that would establish criteria for on-site storage of LLW after January 1, 1996. Licensees would have to exhaust all other reasonable waste management options before storing LLW on-site. Options include contracting, either directly or indirectly, through the State, for disposal.
- Proposed rulemaking would supplement, but not supersede, existing regulatory framework. Conditions of proposed rulemaking, in themselves, would not authorize on-site storage.

Attachment

INCIDENT INVESTIGATION TEAMS

Background:

The Incident Investigation Program (IIP) ensures that significant operational events are investigated by the NRC in a systematic, technically sound, and timely manner, to gather information pertaining to the probable root causes of the event, including any NRC contributions or lapses. The investigation of all significant operational events involving non-reactor and reactor activities licensed by the NRC are within the scope of this program. By focusing its efforts on uncovering the causes of operating events and on identifying associated corrective actions, the incident investigation process provides for a more complete technical and regulatory understanding of significant events. This is a major contribution to nuclear safety. Within the program, appropriate feedback is solicited from the NRC, the nuclear industry, and the public regarding the lessons learned from the events.

Incident Investigation Team:

For an event of high safety significance, an Incident Investigation Team (IIT) is established by the EDO. An IIT performs the single NRC incident investigation and reports directly to the EDO. It is composed of headquarters and regional staff and, for events involving power reactors, the team may also include industry representatives. To maintain independence, the team leader and other technical experts are chosen from those who have had no significant involvement with licensing and inspection activities at the facility at which the event occurred.

For an event of less safety significance, the Regional Administrator appoints an Augmented Inspection Team (AIT). This team reports directly to the Regional Administrator. Its members come from the regional staff supplemented by headquarters personnel and, in some cases, by personnel from other regions.

The team investigations are presently governed by two NRC documents:

(1) Management Directive 8.3, "NRC Incident Investigation Program," and

(2) NUREG-1303, "Incident Investigation Manual."

The IIT begins its investigation as soon as practicable after the facility has been placed in a safe, secure, and stable condition. If there is an NRC incident response, the investigation begins after it is deactivated.

For all IITs a Confirmatory Action Letter (CAL) will be issued as appropriate to the licensee requiring that, within the constraints of plant safety, suspected relevant failed components or systems be quarantined for troubleshooting. The CAL also ensures that the facility is maintained in a safe shutdown condition until concurrence is received from the NRC to restart.

The IIT investigation emphasizes fact-finding and determination of probable causes for the event. As the investigation progresses, the IIT issues interim reports outlining the status and plans, and adds relevant new information related to the investigation. The scope of the investigation is established to ensure that the event is clearly understood, that relevant facts and circumstances are identified and collected, and that probable cause(s) and contributing cause(s) are identified and substantiated by the evidence and information collected. The team also considers whether the licensee and the NRC took timely and adequate action or if they contributed to the cause before or during the event. The team also investigates event precursors, event chronology, systems response, human factors considerations, equipment performance, safety significance, and radiological considerations.

The IIT prepares a final report listing detailed findings and conclusions and sends copies to the Commission and the EDO within about 45 days from the time the team is activated. After receiving the final report, the EDO normally schedules a meeting so the IIT can brief the Commission on the investigation. Information contained in the report is not released to the public until a copy of the report is placed in the Public Document Room (PDR). Following the Commission briefing, a copy of the final report is transmitted to the licensee and NRC staff for their review and comment.

In addition to NRC staff and licensee review and comment on the IIT report, the EDO identifies from the investigation findings generic and plant-specific staff actions for evaluation and resolution as well as actions that are safety significant and that warrant additional attention or action. The EDO assigns these actions to the appropriate NRC office. Office Directors are asked to issue a status report on the disposition of each assigned staff action. AEOD will prepare annual closeout reports to identify and document actions taken to close out each assigned staff action item. In addition, the resolution of each IIT finding is subject to an independent assessment by AEOD of its adequacy and completeness.

<u>Contact</u>:

Stuart Rubin, Chief, Diagnostic Evaluation and Incident Investigation Branch, Division of Operational Assessment, Office for Analysis and Evaluation of Operational Data, U. S. Nuclear Regulatory Commission, Washington, DC 20555, (301) 492-4147.

HIGHLIGHTS OF INCIDENT INVESTIGATION TEAMS

- The Incident Investigation Program (IIP) ensures that NRC investigations of significant events are timely, thorough, well coordinated, and formally administered.

- The scope of the IIP covers the investigation of significant operational events involving non-reactor and reactor activities licensed by the NRC.

- The IIP process contributes to nuclear safety by providing for a more complete technical and regulatory understanding of significant events.

- Appropriate feedback regarding what is learned from the events is provided to the NRC, the industry, and the public.

INCIDENT INVESTIGATION PROGRAM:

- For an event of potentially major significance, an Incident Investigation Team (IIT) is established by the EDO.

- The investigation of less significant operational events may be conducted by an Augmented Inspection Team (AIT).

- The IIT begins its investigation as soon as practicable after the facility has been placed in a safe, secure, and stable condition.

- The IIT issues interim status reports at appropriate intervals.

- The investigation performed by an IIT emphasizes fact-finding and determination of probable causes for a significant operational event.

- The IIT prepares a final report and transmits it to the Commission and the EDO.

- Following NRC staff and licensee review and comment on the IIT report, the EDO identifies and assigns NRC Office responsibility for generic and plant-specific staff actions.

- Each resolution of an IIT finding is subject to independent assessment by AEOD as to its adequacy and completeness.

- The IIT investigation is governed by two NRC Documents:

  (1) Management Directive 8.3, "NRC Incident Investigation Program," and (2) NUREG-1303, "Incident Investigation Manual."

ROSEMOUNT TRANSMITTERS

Background:

During the late 1980's several utilities were having problems with Rosemount transmitters failing to calibrate or failing response time testing. A notification pursuant to 10 CFR Part 21 was issued and NRC started an investigation. It was determined that the problem appeared when the transmitters had a very slow leakage of fill oil. This reduced the travel of the diaphragm within the transmitter and led to failure. On April 21, 1989, the NRC issued Information Notice (IN) 89-42, "Failure of Rosemount Models 1153 and 1154 Transmitters," to alert the industry to the reported failures of Models 1153 and 1154 pressure and differential pressure transmitters manufactured by Rosemount Inc. Rosemount investigated the cause of the failures and confirmed that the failure mode was a gradual loss of fill-oil from the sealed sensing module of the transmitter. At that time it appeared that the failures were lot- related and that "suspect lots" had a higher failure rate than non-suspect lots. On March 9, 1990, the NRC issued Bulletin 90-01. This Bulletin requested that licensees promptly take corrective actions for Model 1153 Series B, Model 1153 Series D, and Model 1154 transmitters manufactured by Rosemount that were in the suspect lots, and therefore, had the potential for leaking fill-oil.

SUPPLEMENT 1 to BULLETIN NO. 90-01:

Since that time, there has been additional information collected and analyzed, resulting in the writing of an updated supplement to the original Bulletin. During the licensee response period to the original Bulletin, NUMARC surveyed all utilities to collect data on all installed Rosemount Model 1153 and 1154 transmitters, and on Rosemount Model 1151 and 1152 transmitters installed in safety-related systems. The staff, assisted by Brookhaven National Laboratory, reviewed the data collected by NUMARC. The failed transmitters were sorted by operating pressure and time-in-service. In evaluating this issue, the staff confirmed a relationship, as had been previously found by Rosemount and NUMARC, between the likelihood of failure and operating pressure, time-in-service, and the suspect and nonsuspect lot classifications. A high operating pressure was the most dominant factor leading to a loss of fill-oil, with time-in-service also being a significant factor. Transmitters with an operating pressure greater than 1500 psi had the highest failure rate, and those with an operating pressure between 1500 and 500 psi also had an increased failure rate. Second among these factors was time-in-service, with those transmitters having been in service for less than 60,000 psi-months exhibiting higher failure rates than transmitters that had been in service for more than 60,000 psi-months.

On April 7, 1992, a proposed Supplement 1 to Bulletin 90-01, was published in the Federal Register. Twelve replies to this notice were received. The comments received primarily concerned the scope of coverage for the transmitters to be addressed and clarification of the exact nature of requested actions. On July 23, 1992, a public meeting was held to discuss the comments received and their disposition. On September 8, 1992, the Committee to Review Generic Requirements (CRGR) reviewed the proposed supplement, including the disposition by the staff of comments received, and recommended changes which the staff incorporated into the supplement. The Bulletin Supplement was issued on December 22, 1992.

The Supplement requests utilities to review the information for applicability to their facilities, perform testing on a transmitter commensurate with its importance to safety and demonstrated failure rate, and modify, as appropriate, their actions and enhanced surveillance programs.

Regulation:

The NRC issued General Design Criterion (GDC) 21, "Protection System Reliability and Testability," in Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) to require the protection system to be designed with high functional reliability and with sufficient capability to allow periodic testing of its functioning when the reactor is operating. The NRC established this requirement to ensure that the licensee can readily detect failures of subcomponents and subsystems within the protection system and can readily detect loss of the required protection system redundancy when it occurs. In 10 CFR 50.55a(h), the NRC requires that protection systems meet the Institute of Electrical and Electronics Engineers Standard, "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE-279). In IEEE-279, it is stated that means shall be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation. To achieve a high functional reliability, a transmitter must have a low probability of failing while it is operating. When failures occur, they should be readily detectable, commensurate with the safety function, while the transmitter is operating. Upon reviewing the analyses, evaluations, and historical data on the loss of fill-oil, the staff concluded that actions requested by the previous bulletin were insufficient to ensure the transmitters achieve the desired high functional reliability.

Current Status:

The Bulletin Supplement, issued on December 22, 1992, also requested that within 60 days after receipt of the bulletin, the licensee provide a response that includes a statement whether the licensee will take the actions requested. This statement includes a list of the specific actions that the licensee will complete and the schedule for completing actions. The licensee is also requested to provide a statement confirming that the requested actions have been completed. If the licensee is not completing all actions, a statement is requested identifying those actions that the licensee is not taking and an evaluation which provides the bases for not taking the requested actions. All responses have been received and review of the responses will continue through 1994. Additionally, a review group, the Rosemount Transmitter Review Group (RTRG), was formed. The RTRG was tasked to perform an in-depth review and evaluation to determine whether the agency should require licensees to take additional action beyond that specified in Bulletin 90-01 and Supplement 1 to Bulletin 90-01. The RTRG completed its evaluation, and issued its report on October 12, 1993. The principal conclusions of the RTRG were that the scope and actions specified in NRC Bulletin 90-01, Supplement 1 are appropriate and that improvements in Rosemount Model 1153 B/D and 1154 transmitters manufactured since July 11, 1989, have significantly reduced the transmitter failure rate. The RTRG, however, also recommended that the following actions be taken: (1) issue a temporary instruction for NRC inspections of the effectiveness of licensee actions in response to Bulletin 90-01, Supplement 1, and collection data on calibration trending and failures of all Rosemount transmitters; (2) continue periodic dialogue with Rosemount to track the performance of the different model transmitters; (3) review Nuclear Plant Reliability Data System data on Rosemount transmitters every six months for two years; (4) hold management meetings with NUMARC to discuss lessons learned from the Rosemount transmitter loss-of-fill-oil issue; (5) review EPRI Report TR-102908 dealing with Rosemount transmitter concerns; and (6) ask the NRC Office of General Counsel to provide a written, legal interpretation regarding the circumstances under which organizations such as NUMARC and EPRI would be subject to the requirements of 10 CFR Part 21 and 10 CFR Part 50.9 for reporting defects and noncompliance. The above actions are being implemented in 1994 and will provide the NRC with additional information to ensure that actions taken in response to NRC Bulletin 90-01, Supplement 1 are sufficient to resolve the Rosemount transmitter concerns.

Technical Contact: Deirdre Spaulding, NRR/HICB (301) 504-2928
Lead Project Manager: Ngoc Le, NRR (301) 504-1458

HARASSMENT AND INTIMIDATION IN THE NUCLEAR INDUSTRY

Background:

The NRC has responsibility for regulating the operation of more than 100 nuclear power plants and more than 8,000 materials licensees. The magnitude of the licensed activities is so extensive that the NRC can inspect only a fraction of them. Although the NRC's program for ensuring adequate protection is not structured to be dependent upon allegations of safety deficiencies, the NRC will never have the knowledge possessed by the thousands of employees in the nuclear industry. The NRC, therefore, expects employees in the nuclear industry to be free to raise potential safety issues.

Recognizing that it is not the NRC, but licensees who have the first responsibility for safety, it is not enough for employees to feel free to come directly to the NRC. Employees must feel free to raise potential safety issues to their management. Over the years the NRC, the regulated industry, and the public have benefitted from the issues raised by employees of licensees and their contractors. If employees are subjected to retaliation by their employers for reporting safety concerns, there is a potential "chilling effect" for both allegers and co-workers who may have additional safety concerns to report.

The current regulatory process seeks to provide protection against retaliation for employees engaged in protected activities, for instance, raising of potential safety concerns to a licensee or the NRC. Discrimination against an employee for engaging in protected activities is prohibited by the Commission's regulations (10 CFR 19.20, 30.7, 40.7, 50.7, 60.9, 61.9, 70.7, and 72.10). Discrimination for purposes of the Commission's regulations includes discharge and other actions that relate to compensation or terms, conditions, and privileges of employment. A licensee is subject to enforcement action under the Atomic Energy Act (AEA) of 1954, as amended, for violations of these prohibitions by itself or its contractors and subcontractors.

The AEA does not give the NRC the authority to provide a personal remedy, such as reinstatement or back pay, to an employee who has been subject to discrimination. The NRC's authority is directed at the licensee. NRC regulations and Federal statutes allow enforcement actions against licensees that could include denying, revoking, or suspending a license; imposing civil penalties; and initiating criminal proceedings.

An employee who believes that discrimination has occurred may seek a personal remedy by filing a complaint within 180 days with the Department of Labor (DOL) in accordance with Section 211 of the Energy Reorganization Act (ERA) of 1974, as amended. However, the DOL process for providing personal remedies for those retaliated against (and the NRC process for taking enforcement action against the licensees who have retaliated) is time consuming.

Although the NRC can initiate an investigation at any time during the DOL process, the agencies agreed in a 1982 Memorandum of Understanding that the NRC would not normally conduct a parallel investigation. To a large degree, the NRC relies on the DOL for investigating allegations of discrimination. However, the NRC does respond to technical aspects of allegations.

Between October 1988 and April 1993, the NRC received a total of 609 retaliation complaints and initiated full-scale investigations for 44 of them. Three hundred-sixty nine of these complaints were also filed with DOL. Based on the complaints received, seven NRC enforcement actions were taken against licensees during this period and others are pending.

On July 9, 1993, the NRC Inspector General (IG) issued a report that addressed the staff's handling of retaliation complaints received from whistleblowers employed by NRC licensees. The IG, after interviewing various whistleblowers and NRC staff members, concluded that the NRC process for handling allegations of retaliation does not provide an adequate level of protection for whistleblowers.

On July 15, 1993, the NRC testified in a hearing conducted by the Senate Subcommittee on Clean Air and Nuclear Regulation on the NRC's handling of intimidation and harassment allegations by employees within the nuclear industry.

Current Developments:

On July 6, 1993, the Executive Director for Operations established a Review Team for Reassessment of the NRC Program for Protecting Allegers Against Retaliation. The Review Team is to determine whether the Commission has taken sufficient steps within its authority to create an atmosphere within the regulated community where individuals with safety concerns feel free to engage in protected activities without fear of retaliation. Because Section 211 of the ERA was recently amended as part of the Energy Policy Act of 1992, Pub. L. 102-486, this review is to focus primarily on the existing statutory framework, in which the DOL provides the personal remedy to the employee, and NRC is responsible for regulating licensees such that licensees will foster an atmosphere where individuals will be encouraged to come forward with safety information without fear of retaliation.

In accordance with its charter, the Review Team is to consider:

(a) Whether the NRC has taken sufficient action through issuance of regulations, policy statements, and inspections to ensure that NRC licensees encourage their employees and contractors to raise safety concerns without fear of reprisal;

(b) Whether the current NRC process for handling allegations is appropriate from the perspective of allegers' feeling free to bring safety concerns to the NRC; and

(c) Where discrimination may have occurred --

(1) Whether there are NRC actions that can assist in a speedier resolution of issues within the DOL process;
(2) Whether NRC should be more proactive in conducting investigations during the pendency of DOL proceedings;
(3) Whether the NRC takes sufficient follow up action to determine if the licensee has taken action to remove the potential chilling effect arising from the discrimination;
(4) Whether the NRC can and should use civil penalties and orders more vigorously to emphasize the need for licensees actively to encourage employees to raise safety concerns without fear of discrimination; and
(5) Whether the NRC can and should use orders and demands for information more vigorously, where individuals are found to have caused discrimination; and

(d) Whether the NRC is sufficiently proactive in cases where employees raise concerns with the NRC and express fear that they may become subject to retaliation for raising safety concerns.

During September and October 1993, the Review Team conducted public meetings with licensees and their employees and other interested individuals to obtain views on whether the NRC has taken sufficient steps to create the right atmosphere at its licensed facilities. In accordance with its revised charter, the Review Team's report is due to the Commission by January 14, 1994.

On October 8, 1993, the NRC published a final rule, revising appropriate sections of 10 CFR Parts 19, 30, 40, 50, 60, 61, 70, 72, and 150, due to recent amendments to Section 210 of the Energy Reorganization Act of 1974. These amendments changed the number of the section from 210 to 211, extended the period for whistleblowers to file a complaint with the Department of Labor from 30 days to 180 days, and extended and/or clarified protection to new classes of employees and employers.

Contact:

James Lieberman, Director, Office of Enforcement
US NRC, Washington, DC 20555. (301) 504-2741.

HIGHLIGHTS OF H & I ISSUES IN THE NUCLEAR INDUSTRY

- NRC expects employees in nuclear industry to be free to raise potential safety concerns.
- If employees are subjected to retaliation by their employers for reporting safety concerns, there is a potential "chilling effect" for both whistleblowers and co-workers who may have additional safety concerns to report.
- Discrimination against an employee for engaging in protected activities is prohibited by the Commission's regulations (10 CFR 19.20, 30.7, 40.7, 50.7, 60.9, 61.9, 70.7, and 72.10).
- Discrimination for purposes of the Commission's regulations includes discharge and other actions that relate to compensation or terms, conditions, and privileges of employment.
- AEA doesn't give NRC authority to provide personal remedy (reinstatement or back pay). NRC's authority is directed at licensee--NRC could deny, revoke, or suspend a license; impose civil penalties; and refer to DOJ for criminal proceedings.
- DOL has authority to provide personal remedy, when appropriate--but process has been time consuming.
- In accordance with 1982 MOU with DOL, agencies agreed that NRC would not normally conduct parallel investigation. While NRC responds to technical aspects of allegations, NRC, to a large extent, relies on the DOL for investigating allegations of discrimination.
- NRC IG issued report on July 9, 1993, that concluded that the NRC process for handling allegations of retaliation does not provide an adequate level of protection for whistleblowers.
- NRC testified in hearing on July 15, 1993, to the Senate Subcommittee on Clean Air and Nuclear Regulation on the NRC's handling of intimidation and harassment allegations by employees within the nuclear industry.
- EDO established Review Team on July 6, 1993, for reassessment of the NRC program for protecting allegers against retaliation.

DECOMMISSIONING NUCLEAR POWER PLANTS

Background:

Recently, several licensees announced their decisions to permanently cease power operation at their nuclear power generating facilities. The reasons for their decisions include both economic and technical considerations. Thus, these facilities and several others entered the decommissioning process earlier than originally anticipated. Recent decommissioning highlights are prese[illegible] in Table 1 and a status for decommissioning plan[illegible] presented in Table 2.

Decommissioning:

Title 10 of the Code of Federal Regulations, Section 50.2 (10 CFR 50.2), defines decommissioning to mean removal of a facility safely from service, reduction [illegible]esidual radioactivity to a level that permits release of [illegible]operty for unrestricted use, and termination of the license. Decommissioning can involve three different methods: DECON, SAFSTOR, or ENTOMB. Under DECON, equipment, structures and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use relatively soon (a few years) after cessation of operations. Under SAFSTOR, often considered "delayed DECON," a nuclear facility is placed and maintained in a condition that allows safe storage for a per[illegible] of time to allow radioactive decay to reduce radiation leve[illegible] at the facility. The plant is subsequently decontaminated to levels that permit release of the site for unrestricted use. Under ENTOMB, radioactive contaminants are encased in a structurally long-lived material such as concrete and the entombed structure is appropriately maintained and monitored until the radioactivity decays to a level permitting unrestricted release of the property. The ENTOMB alternative has generally not been selected by nuclear power plant licensees. These three decommissioning alternatives are not totally discrete actions inasmuch as some decontamination and other preparatory activities such as component disassembly may be performed under both the SAFSTOR and ENTOMB alternatives. Finally, to be acceptable, the method selected must provide for completion of decommissioning within 60 years and a time beyond 60 years will be considered only when necessary to protect public health and safety, in accordance with NRC regulations.

Regulations:

The procedure for decommissioning a nuclear power plant is set out principally in 10 CFR 50.75, 50.82, 51.53, and 51.95. (An underlying, assumption embodied in the current regulations is that decommissioning would occur after the time of expiration of the facility operating license.) The formal process begins with the filing of an application by the licensee, normally after the plant has ceased permanent operations, for authority to decommission the facility. Five years before the licensee expects to end operation of the plant, the licensee is obligated to submit a preliminary decommissioning plan containing a cost estimate for decommissioning and an up-to-date assessment of the major technical factors that could affect planning for decommissioning. Then, within two years following permanent cessation of operations, but no later than one year prior to expiration of the license, a licensee must submit to the NRC an application for authority to decommission that facility, together with an environmental report covering the proposed decommissioning activities. The application must also be accompanied, or preceded, by a proposed decommissioning plan that includes:

(1) A description of the decommissioning alternative chosen and activities involved.

(2) A financial plan showing a cost estimate for decommissioning, the amount of funds currently available for decommissioning, and plans for assuring the availability of adequate funds for completion of decommissioning.

The NRC reviews the decommissioning plan, prepares an environmental impact statement (EIS) or environmental assessment (EA), as appropriate, and gives notice to interested persons. If the NRC finds the proposed decommissioning plan to be satisfactory, the NRC issues a decommissioning order that approves the proposed decommissioning plan and authorizes decommissioning. Upon completion of decommissioning activities, including the completion of a termination radiation survey, the NRC will issue an order that terminates the license.

Prematurely Shutdown Plants:

Subsequent to the publication of the final decommissioning rule six power reactor facilities have been shut down prematurely: the Fort St. Vrain Nuclear Generating Station, the Shoreham Nuclear Power Station, the Rancho Seco Nuclear Generating Station, the Yankee Rowe Nuclear Station, San Onofre Nuclear Generating Station, Unit 1, and Trojan Nuclear Plant. Three Mile Island Nuclear Station, Unit 2 also ceased operation following the March 28, 1979, accident.

Current Status:

During CY-1992, the NRC completed the review efforts necessary to support decommissioning of the Shoreham and Fort St. Vrain facilities. On June 11, 1992, the NRC issued an order to the current Shoreham licensee, Long Island Power Authority, approving the Shoreham decommissioning plan. The Long Island Power Authority is in the process of dismantling that facility. On November 23, 1992, the NRC issued an order approving the Fort St. Vrain decommissioning plan and dismantlement activities are now ongoing. Issuance of the Rancho Seco decommissioning plan has been delayed due to the Commission remanding to the Atomic Safety Licensing Board three issues (loss of offsite power, decommissioning funding plan, and decommissioning environmental assessment) raised by the Environmental and Resources Conservation Organization (ECO). On June 16, 1993, the NRC staff issued its safety evaluation and its environmental assessment on the proposed Rancho Seco decommissioning plan. This matter is now before the Licensing Board. On November 30, 1993, the ASLB denied, in their entirety, admission of the Loss of Offsite Power and environmental assessment contentions. While some decommissioning funding contentions were also denied, the ASLB admitted for hearing contentions associated with material aspects of decommissioning funding and costs associated with the Rancho Seco independent spent fuel storage installation.

During fiscal year 1992, Yankee Atomic Electric Company and Southern California Edison Company announced their decisions to prematurely shut down and decommission the Yankee-Rowe and San Onofre 1 facilities, respectively. In January 1993, the Portland General Electric Company decided to terminate operations at the Trojan plant. All three of these facilities now have been permanently shut down. The NRC is working to support the decommissioning of these facilities.

In January 1993, the Commission issued guidance regarding activities which may be permitted prior to approval of a decommissioning plan. Licensees of plants which have possession only licenses or shutdown orders should be allowed to undertake any decommissioning activity that does not (a) foreclose the release of the site for possible unrestricted use, (b) significantly increase decommissioning costs, (c) cause any significant environmental impact not previously reviewed, or (d) violate the terms of the existing license. Also, licensees may be permitted to use their decommissioning funds for approved decommissioning activities, notwithstanding the fact that their decommissioning plans have not yet been approved by the NRC. In accordance with this guidance, Yankee Atomic Electric Company is conducting early removal of the four steam generators, the pressurizer, and reactor vessel internals from the Yankee Rowe plant. In a July 15, 1993, letter to Yankee, the staff stated that it has no objection to these activities.

Contact:

Seymour H. Weiss, Non-Power Reactors & Decommissioning Project, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, (301) 504-2170.

TABLE 1

DECOMMISSIONING HIGHLIGHTS

FORT ST. VRAIN

- August 18, 1989, plant was permanently shut down because of failure of the control rod drives and degradation of the steam generator ring header.
- May 21, 1991, license is amended to a POL status.
- June 11, 1992, all fuel placed in an onsite independent spent fuel storage installation (ISFSI).
- November 23, 1992, the Commission issued an order approving licensee decommissioning plan.
- September 1, 1993, the Prestressed Concrete Reactor Vessel top head has been successfully removed.
- December 8, 1993, approximately one-half of the graphite reflector blocks have been removed from the reactor vessel and shipped to the low level waste burial site in Richland, WA.

SHOREHAM

- June 28, 1989, licensee's shareholders approve an agreement with the New York state to not operate the facility.
- August 24, 1989, reactor vessel defueled.
- June 14, 1991, license is amended to a POL status.
- February 29, 1992, the license was transferred to the Long Island Power Authority for decommissioning.
- June 11, 1992, Commission issued an order approving licensee decommissioning plan.
- September 1, 1993, decommissioning of the facility is 75 percent complete.
- September 1993, transfer of fuel to Limerick began. Fuel transfer is to be completed by September 1994.

RANCHO SECO

- June 7, 1989, plant shut down as a result of approval by voters of a non-binding referendum prohibiting the licensee from operating the facility.
- December 8, 1989, reactor vessel defueled.
- March 17, 1992, license is amended to a POL status.
- Active intervention on proposed decommissioning plan by Environmental and Resources Conservation Organization (ECO).
- June 3, 1993, U.S. Court of Appeals (Ninth Circuit) rules in favor of the NRC in the matter of the issuance of the Ranch Seco POL.
- June 16, 1993, safety evaluation and environmental assessment of proposed decommissioning plan issued by NRC staff.

- November 30, 1993, ASLB denied admission of Loss of Offsite Power and environmental assessment contentions. The ASLB admitted for hearing contentions associated with material aspects of decommissioning funding and costs associated with the Rancho Seco independent spent fuel storage installation.

YANKEE ROWE

- February 27, 1992, the licensee announced permanent cessation of operations because of inability to address uncertainties associated with the safety margin of the reactor vessel. The vessel was previously defueled.
- August 5, 1992, license is amended to a POL status.
- July 15, 1993, NRC staff states "no objection to early component removal activities."
- The four steam generators and pressurizer were shipped from the plant to the low level burial site in Barnwell, SC, between November 16 and December 8, 1993.

THREE MILE ISLAND, UNIT 2

- March 28, 1979, accident occurred in the plant that caused permanent cessation of operations.
- January 30, 1990, reactor was defueled.
- August 12, 1993, processing of Accident-Generated Water is completed.
- September 14, 1993, a POL license amendment was issued.

SAN ONOFRE, UNIT 1

- November 30, 1992, plant was permanently shut down rather than bring it into compliance with current NRC safety requirements.
- October 23, 1992, a POL license amendment was issued. POL amendment became effective March 9, 1993, when the reactor vessel was certified as completely defueled.

TROJAN

- January 4, 1993, the licensee announced permanent cessation of operations.
- January 27, 1993, reactor was defueled.
- May 5, 1993, a POL license amendment was issued.

TABLE 2

REACTOR DECOMMISSIONING STATUS
SHUTDOWN POWER REACTORS

DOCKET NO. REACTOR	THERMAL POWER	LOCATION	SHUT DOWN	PRESENT STATUS	FUEL ONSITE?
50-3 Indian Point-1 (PWR)	615 MW	Buchanan New York	10/31/74	Possession Only Lic.	Yes
50-10 Dresden 1 (BWR)	700 MW	Morris Illinois	10/31/78	SAFSTOR Approved	Yes
50-16 Fermi 1 (Fast Breeder) *	200 MW	Monroe Co. Michigan	9/22/72	SAFSTOR Approved	No
50-18 GE VBWR (BWR) *	50 MW	Alameda Co. California	12/9/63	SAFSTOR Approved	No
50-29 Yankee Rowe (PWR)	600 MW	Franklin Co. Massachusetts	10/1/91	Possession Only Lic.	Yes
50-114 CVTR (Pressure Tube, Heavywater)	65 MW	Parr S. Carolina	Jan. 67	Byproduct Lic. (St.)	No
50-130 Pathfinder (Nuclear Superheat BWR) *	190 MW	Sioux Falls S.D.	9/16/67	DECON NRC Part 30	No
50-133 Humboldt Bay-3 (BWR) *	200 MW	Eureka California	7/2/76	SAFSTOR Approved	Yes
50-171 Peach Bottom 1 (HTGR) *	115 MW	York Co. Pennsylvania	10/31/74	SAFSTOR Approved	No
50-206 San Onofre 1 (PWR)	1347 MW	San Clemente California	11/30/92	Possession Only Lic.	Yes
50-267 Fort St. Vrain (HTGR) *	842 MW	Platteville Colorado	8/18/89	DECON Approved	Yes
50-312 Rancho Seco (PWR)	2772 MW	Sacramento California	6/7/89	Possession Only Lic.	Yes
50-320 Three Mile Island 2 (PWR)	2772 MW	Middletown Pennsylvania	3/28/79	Shutdown Defueled	No
50-322 Shoreham (BWR) *	2436 MW	Suffolk Co. New York	6/28/89	DECON Approved	Yes
50-344 Trojan (PWR)	3411 MW	Portland Oregon	11/9/92	Possession Only Lic.	Yes
50-409 LaCrosse (BWR) *	165 MW	LaCrosse Wisconsin	4/30/87	SAFSTOR Approved	Yes

* Project management assigned to NMSS

BWR WATER LEVEL INSTRUMENT ERRORS DUE TO NONCONDENSIBLE GAS

Background:

Level anomalies have been observed in reactor vessel water level indication at several boiling water reactors (BWRs) during controlled depressurization while commencing plant outages or following reactor trips. These anomalies consisted of "spiking" or "notching" of level indication, and in one instance, a sustained error in level indication. The root cause of these level indication anomalies has been determined to be the effects of non-condensible gas dissolved in the reference leg of "cold reference leg" type water level instruments. Under rapid depressurization conditions non-condensible gases could cause significant errors in the level indication.

Discussion of Technical Issue:

Cold reference leg water level instruments measure reactor vessel water level by measuring the differential pressure of two columns of water, the variable leg and the constant height reference leg. The reference leg is maintained filled to a constant height of water by the condensate chamber. Steam is condensed in the condensate chamber and keeps the reference leg full. Excess condensate is returned to the vessel through the steam supply line. Non-condensible gases, such as hydrogen and oxygen, formed by radiolysis in the reactor vessel, are present in the steam supplied to the condensate chamber. The gases can collect in the condensate chamber and can accumulate to high partial pressures. The gases then become dissolved in the water at the top of the reference leg, and the dissolved gases can be transported down the reference leg by small leaks in valves and fittings at the bottom of the reference leg, diffusion, and/or thermal convection.

Dissolved gases in the reference leg do not present a problem unless the instrument is depressurized. When depressurized, the gases come out of solution and form bubbles that travel up the reference leg. During slow depressurization, level indication has been seen to temporarily "spike" or "notch" while a bubble moves through the vertical sections of the piping. Significant spiking may automatically actuate such systems as the primary containment isolation system (PCIS); this occurred at the Pilgrim plant. After spiking, which is of short duration, the indicated water level returns to actual level. Level spiking is of lesser safety significance. Bubbling of the gases may eject a significant amount of water from the reference leg. Loss of reference leg inventory will cause level indication to be erroneously high. This occurred during a normal plant cooldown on January 21, 1993 at WNP-2, resulting in a 32 inch error in level indication that gradually recovered over a period of 2 hours. If the reactor is rapidly depressurized, as would occur during a design-basis loss-of-coolant accident (LOCA) or opening of the automatic depressurization system (ADS) valves, even larger errors in the level indication could result. However, analyses presented by the industry indicated that significant errors would not be expected until the reactor is depressurized below approximately 450 psi.

The staff has taken several actions to address this problem. The BWR Owners Group (BWROG) Regulatory Response Group (RRG) was activated during July 1992. The staff also issued Information Notice 92-54 in July 1992, Generic Letter 92-04 in August 1992, and Information Notice 93-27 in March 1993 to alert licensees to the potential problem and to request information concerning actions taken or planned by licensees in response to potential errors in level indication. The BWROG conducted a test program to support their efforts to resolve this issue. The results of the BWROG reference leg de-gas test program confirmed that no significant errors in level indication will occur until the reactor is depressurized below 450 psig, and that large errors in level indication are possible once the reactor is depressurized to lower pressures.

The staff received additional information from the BWROG pertaining to reactor vessel water level instrumentation inaccuracies during normal depressurization due to the effects of noncondensible gas. At the staff's request the BWROG submitted a report on May 20, 1993, discussing the impact of level errors on automatic safety system response and operator actions during transients and accidents initiated from reduced pressure conditions during plant cooldown (Mode 3). Based on this information, in addition to the January 21, 1993, WNP-2 event and data from the reference leg de-gas testing that was conducted by the BWROG, the staff concluded that additional short-term actions needed to be taken for protection against potential events occurring during normal cooldown. On May 28, 1993, NRC Bulletin (NRCB) 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation," was issued, in which the staff requested each BWR licensee to implement additional short term compensatory actions, and to implement a hardware modification to resolve this issue at the next cold shutdown after July 30, 1993.

The staff has received responses to NRC Bulletin 93-03 from all licensees. All licensees have completed the short term compensatory actions and have committed to install hardware modifications. Delays in implementation were granted to some licensees to allow sufficient time to complete necessary design and hardware procurement to support the installation of hardware modifications. The majority of licensees have decided to install a backfill modification which would constantly purge the reference leg with a very low flow rate (0.008 gpm) of water supplied by the Control Rod Drive (CRD) system. The constant flow of water up the reference leg would prevent dissolved gases from migrating down the reference leg. At this time, several plants have completed installation of backfill modifications. Remaining licensees plan to complete modifications during the next planned refueling outage, or if an unplanned shutdown should occur, they plan to install the modification at that time.

Current Status:

On November 26, 1993, the staff issued Information Notice (IN) 93-89, "Potential Problems with BWR Level Instrumentation Backfill Modifications," to alert licensees to potential problems that have been identified involving hardware modifications to the reactor vessel water level instrumentation system. This information involved the potential to pressurize the reference legs of the water level instrumentation if a reference leg backfill system is installed with the injection point on the instrumentation side of the manual isolation valve in the reference leg, and if that valve is closed inadvertently during backfill system operation. Inadvertent closure of this valve could result in a severe plant transient. The consequences would vary significantly with plant specific instrumentation systems. At some plants, as discussed in the IN, valve closure would cause all safety relief valves to open and potentially impact emergency core cooling system response. Some licensees have determined that administrative controls are acceptable to prevent valve closure, others have chosen to implement design features to prevent this scenario, such as injecting on the reactor side of the manual isolation valve. The staff is reviewing the acceptability of administrative controls to prevent this scenario from occurring.

Contact:

Amy E. Cubbage, BWR Reactor Systems Section, Reactor Systems Branch, Division of Systems Safety and Analysis, US NRC, Washington, DC 20555. (301) 504-2875

HIGHLIGHTS OF BWR WATER LEVEL INSTRUMENT ERRORS DUE TO NONCONDENSIBLE GAS

- Potential for significant errors in level indication when reactor is rapidly depressurized below approximately 450 psig, and during normal plant depressurization below 450 psig.

- Information Notice No. 92-54, Generic Letter No. 92-04, Information Notice No. 93-27, NRC Bulletin No. 93-03, and Information Notice No. 93-89 were issued by the staff.

- Staff has concluded that interim plant operation is acceptable.

- In the long-term, however, the staff expects each licensee to implement hardware modifications to ensure that its level instrumentation system design is of high functional reliability.

- All affected licensees have committed to install hardware modifications to resolve this issue.

FIRE BARRIER ISSUES

Regulatory Background:

NRC-approved plant fire protection programs as referenced by the Plant Operating License Conditions and Appendix R to 10 CFR Part 50, Section III G.1.a, "Fire Protection of Safe Shutdown Capability," require one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control stations to be free from fire damage.

To ensure that electrical cables and components are free from fire damage, Section III G.2 of Appendix R requires that the safe shutdown trains be separated by one of the following:
(1) separating cables and equipment and associated circuits of redundant trains by a fire barrier having a 3-hour rating,
(2) separating cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazard, or
(3) enclosing cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. If the licensee chooses to rely on separation or to install a 1-hour barrier, it must also install fire detection and automatic fire suppression systems in the fire area.

Thermo-Lag Fire barriers:

Approximately 80 licensees use Thermo-Lag fire barriers to satisfy the NRC's fire protection requirements. Thermo-Lag 330-1 is a proprietary fire barrier material manufactured and supplied by Thermal Science, Incorporated. The vendor manufactures Thermo-Lag in nominal 1/2-inch and 1-inch thicknesses which provide fire endurance ratings of 1 hour and 3 hours, respectively.

Thermo-Lag is a sacrificial subliming material that is consumed when it is exposed to a fire. In a fire, the solid material sublimes, the subliming gases are decomposed by the fire, and the virgin Thermo-Lag material is replaced by a char layer. The sublimation process and the insulating effects of the resulting char layer protect the equipment located within the confines of the fire barrier from the effects of the fire. Conversely, more traditional fire barriers, such as concrete block walls, provide fire endurance by maintaining structural integrity during the fire exposure and limiting heat transfer through the barrier.

Staff Concerns:

The staff has three principal concerns with the Thermo-Lag fire barriers: fire endurance, combustibility, and ampacity derating.

Qualification fire tests of cable tray and conduit barriers conducted by industry and small-scale panel tests performed by the NRC staff at the National Institute of Standards and Technology (NIST) demonstrated that Thermo-Lag fire barriers may not provide the fire endurance needed to satisfy the NRC's requirements. The test results led to the issuance of NRC Information Notice (IN) 92-55, NRC Bulletin (NRCB) 92-01, and NRCB 92-01, Supplement 1. The NRC is also concerned that some Thermo-Lag barriers used by some licensees, such as walls and ceilings, have not been qualified as fire barriers by test.

The NRC is also concerned that Thermo-Lag may burn more readily when exposed to fire than originally believed by the NRC and licensees. The staff is concerned about the following possible situations (1) a licensee has not considered the combustibility of the Thermo-Lag materials in its fire hazards analyses, or (2) a licensee is using Thermo-Lag where a noncombustible material should be used. The NRC issued IN 92-82 on December 15, 1992, to inform the licensees of this issue. The staff is considering long-range actions.

Finally, the NRC is concerned that the ampacity derating factors used by the licensees to derate their power cables may not be great enough to actually account for the insulating effects of the Thermo-Lag material. Therefore, cable temperatures may exceed their temperature ratings, possibly resulting in an accelerated aging of the cable insulation.

While the staff is concerned that Thermo-Lag fire barriers may not, in all cases, meet NRC requirements, it believes the relative safety significance of these concerns is low for several reasons, including:

(1) In response to generic communications, licensees have established fire watches to compensate for possibly inoperable fire barriers. This measure provides an adequate level of fire protection until licensees develop and implement permanent corrective actions.

(2) Licensees rely on a defense-in-depth concept where multiple safety measures are incorporated. Automatic fire detection and sprinkler systems are provided in areas which have safe shutdown equipment. Trained fire brigades are required 24 hours per day at all plants. Fuels that can feed a fire and ignition sources to start a fire are controlled. Because of these measures, it is unlikely that a fire significant enough to challenge a fire barrier will occur.

Actions Taken by the Staff:

In June 1991, NRR established a special team to review the safety significance and generic applicability of the technical issues regarding the use of Thermo-Lag. The special review team issued its final report, which identified specific concerns along with their technical bases, in April 1992. Subsequently, the staff prepared an action plan to address the issues associated with Thermo-Lag and the NRC fire protection program. The scope of the action plan includes coordination with industry and testing by the staff.

The staff has issued the following generic communication documents on Thermo-Lag:

- IN 91-47, "Failure of Thermo-Lag Fire Barrier Material To Pass Fire Endurance Test," August 6, 1991.
- IN 91-79, "Deficiencies in the Procedures for Installing Thermo-Lag Fire Barrier Materials," December 6, 1991.
- IN 92-46, "Thermo-Lag Fire Barrier Material Special Review Team Final Findings, Current Fire Endurance Tests, and Ampacity Calculation Errors," June 23, 1992.
- NRCB 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Fire Damage," June 24, 1992, and reviewed licensee responses.
- IN 92-55, "Current Fire Endurance Test Results for Thermo-Lag Fire Barrier Material," July 27, 1992.
- NRCB 92-01, Supplement 1, "Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function," August 28, 1992.
- Generic Letter (GL) 92-08, "Thermo-Lag 330-1 Fire Barriers," December 11, 1992.
- IN 92-82, "Results of Thermo-Lag 330-1 Combustibility Testing," December 15, 1992.

In addition, the staff revised previously developed fire barrier test acceptance criteria to clarify them and incorporate current technical information. The staff provided the improved criteria, which will be applicable to all new fire barrier testing, to the Nuclear Management and Resources Council (NUMARC) on November 19, 1992. The staff will provide these criteria to licensees in a GL, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area (Supplement 1 to GL 86-10, 'Implementation of Fire Protection Requirements')". The proposed GL supplement was published in the Federal Register to solicit public comments on July 23, 1993. The acceptance criteria are near completion.

The staff completed a small-scale fire test program at NIST and witnessed full-scale fire endurance qualification tests and ampacity derating tests performed by Texas Utilities Electric Company for the Comanche Peak Steam Electric Station and by the Tennessee Valley Authority for the Watts Bar Nuclear Plant at Omega Point Laboratories (OPL) in San Antonio, Texas.

NUMARC began constructing specimens for the generic industry testing of Thermo-Lag installations at OPL in July 1993. The test program includes 2 phases. Phase 1 included 6 upgraded test configurations which were funded by Thermal Science, Inc. (TSI, the vendor). NUMARC presented the results of the 6 tests during a Commission meeting on November 24, 1993. Phase 2, which is currently underway, includes 10 test configurations (existing industry installations and additional upgrades) which are funded by NUMARC. The staff is monitoring NUMARC's test program closely by observing activities at OPL and communicating frequently through letters and meetings. The staff has requested additional information pursuant to 10 CFR 50.54(f) from the licensees that use Thermo-Lag fire barriers.

During February 1993, the staff completed a reassement of the NRC fire protection review and inspection programs. The staff has prepared a Fire Protection Task Action Plan to implement the recommendations made as a result of the fire protection program reassessment and is working to resolve the tasks in a timely manner.

The staff is also evaluating fire barriers other than Thermo-Lag. To date, the staff has sent a generic set of questions to all five vendors, other than TSI, believed to supply fire barriers to commercial nuclear power plants. The staff is also reviewing available vendor and licensee test reports and other documents in an effort to verify the adequacy of these fire barriers. Additionally, the staff is in the process of testing these other fire barriers at the National Institute of Standards and Technology.

Based on information obtained to date, the staff has issued the following generic communication documents on these other fire barriers:

- IN 93-40, "Fire Endurance Test Results for Thermal Ceramics FP-60 Fire Barrier Material," May 26, 1993.

- IN 93-41, "One Hour Fire Endurance Test Results for

Thermal Ceramics Kaowool, 3M Company FS-195 and 3M Company Interam E-50 Fire Barrier Systems," May 28, 1993.

Actions Planned by the Staff:

The NRC staff plans to issue the supplement to GL 86-10 as soon as public comments are resolved. The staff will also continue to work with NUMARC and individual licensees to resolve the generic and plant specific technical concerns. The staff will complete its evaluation of the other fire barrier systems used by the licensees to satisfy the NRC's fire protection requirements. A supplement to Generic Letter 92-08 that will request licensees to address issues relating to other fire barrier systems is being considered for issuance in 1994.

Contact:

K. Steven West, Chief, Special Projects Section
Plant Systems Branch, US NRC Washington, DC 20555.
(301) 504-1220

HIGHLIGHTS OF FIRE BARRIER ISSUES

- Appendix R requires separation of safe shutdown trains by a 3-hour fire barrier or a 1-hour barrier with fire detection and suppression.

- Approximately 80 licensees use Thermo-Lag 330-1 fire barriers to comply with Appendix R.

- NRC staff has concerns regarding the fire endurance, combustibility, and ampacity derating of the Thermo-Lag barriers.

- Applicable generic communications on Thermo-Lag include:

  IN 91-47 - Failure of Thermo-Lag Fire Barrier Material To Pass Fire Endurance Test (8/6/91)

  IN 91-79 - Deficiencies in the Procedures for Installing Thermo-Lag Fire Barrier Materials (12/6/91)

  IN 92-46 - Thermo-Lag Fire Barrier Material Special Review Team Final Findings, Current Fire Endurance Tests, and Ampacity Calculation Errors (6/23/92)

  NRCB 92-01 - Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Fire Damage (6/24/92)

  IN 92-55 - Current Fire Endurance Test Results for Thermo-Lag Fire Barrier Material (7/27/92)

  NRCB 92-01, Sup. 1 - Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function (8/28/92)

  GL 92-08 - Thermo-Lag 330-1 Fire Barriers (12/11/92)

  IN 92-82 - Results of Thermo-Lag 330-1 Combustibility Testing (12/15/92)

- NRC staff has published a proposed generic letter supplement (GL 86-10, Supplement 1) that provides fire endurance test acceptance criteria for public comment.

- NRC staff is reviewing other fire barriers used by licensees to satisfy NRC fire protection requirements. Applicable generic communications include:

  IN 93-40 - Fire Endurance Test Results for Thermal Ceramics FP-60 Fire Barrier Material (5/26/93)

IN 93-41 - One Hour Fire Endurance Test Results for Thermal Ceramics Kaowool, 3M Company FS-195 and 3M Company Interam E-50 Fire Barrier Systems (5/28/93)

- NRC staff plans to issue a generic letter supplement to include fire barriers manufactured by other vendors.

STEAM GENERATOR TUBE PLUGGING CRITERIA

Background:

The thin-walled tubing of the steam generator (SG) constitutes over 50% of the reactor coolant pressure boundary (RCPB) for Pressurized Water Reactors. The integrity of this boundary is particularly important in minimizing the release of radioactive fission products to the environment. The Nuclear Regulatory Commission (NRC) has a regulatory program designed to ensure that the structural and leakage integrity of the steam generator tubing is maintained at nuclear power plants in the U.S.

In the early to mid 1970s when phosphate water chemistry was the primary method used for secondary water chemistry control, the main cause of removal from service of SG tubes by plugging was wastage. Wastage is characterized by large area thinning of the tube wall. Therefore, SG tube plugging criteria for most plants were based on uniform thinning of the steam generator tube walls. However, the current dominant degradation mechanism is stress corrosion cracking (SCC).

Tube Plugging Criteria:

Because of the potential consequences of the loss of SG tube integrity, the agency has measures for ensuring that the integrity of the SG tubing is maintained. The traditional tube plugging criteria have typically been based on a minimum wall thickness requirement which assumes that the degradation involves uniform thinning of the tube wall in the axial and circumferential directions. The assumption of uniform thinning conservatively bounds the degrading effects of all flaw types occurring in the field, and is the basis of the standard 40% depth-based plugging limit. However, the 40% plugging limit is very conservative for highly localized flaws such as pits and short cracks. The staff has approved higher depth-based limits, ranging up to 64%, for specific types of flaws at specific plants, for instance, for pitting at Indian Point Unit 3 in the early 1980s.

In recent years, utilities have proposed various flaw-specific plugging limits which do not incorporate a minimum wall thickness requirement. These proposals are often referred to as alternate plugging criteria (APC) since they would be an alternative to the depth-based criterion. Each of these proposals would permit tubes with up to 100% through-wall cracks to remain in service, subject to certain restrictions. These restrictions are intended to ensure adequate structural and leakage integrity. Certain of these proposed limits, known as F* (F-star) and L* (L-star) limits, have been approved by the NRC and have been in use for several years. The F* and L* limits apply to primary water stress corrosion cracking (PWSCC) flaws located within the thickness of the tubesheet for tubes which have been hardroll expanded. Rupture of the tubing within the tubesheet region is precluded by the constraint against tube wall deformation provided by the tubesheet. The specified F* and L* limits have been set to preclude tube pullout from the tubesheet during postulated accidents and to ensure the leaktight integrity between the fully expanded tubing and the tubesheet.

Another type of flaw-specific plugging limit which has been proposed by various utilities involves the use of a bobbin voltage amplitude based limit for axially oriented outside diameter stress corrosion cracking (ODSCC) confined within the tube support plate (TSP) intersections. This proposal includes commitments to the use of enhanced inspection methods, enhanced sampling plans, and reduced primary-to-secondary leak rate limits. The voltage-based plugging limit is established from a burst pressure/voltage amplitude correlation which is adjusted for allowances in voltage amplitude uncertainty and for projected voltage growth during the next operating cycle. This limit is intended to ensure adequate structural and leakage integrity of the tubing.

Another flaw-specific plugging limit includes length-based limits for PWSCC at roll transition locations, similar to limits currently being implemented in some European countries. As is the case for voltage amplitude based limits, proposals for length-based limits are expected to be programmatic, involving commitments to specific inspection methods, inspection sampling plans, and reduced primary-to-secondary leak rate limits as well as revised plugging limits. The Electric Power Research Institute (EPRI) has issued a technical support document to support length-based plugging limit proposals from U.S. utilities.

Current Status:

The NRC staff has approved the use of restricted versions of the voltage-amplitude based limits for axially oriented ODSCC. The restricted versions incorporated much smaller voltage limits than those proposed and limited the use to one operating cycle for several plants (Farley Units 1 and 2, D.C. Cook Unit 1, and Catawba, Unit 1). This restricted version of the APC is referred to as the Interim Plugging Criteria (IPC).

A special task force has reviewed the technical bases for the voltage based IPC. The conclusions and recommendations of the Task Group have been issued as a draft report for comment, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes." The public comments are currently under evaluation and will be incorporated in the final report. The staff expects to issue this report as a generic letter.

The staff is considering a generic approach to defect-specific steam generator inspection and repair criteria. This generic approach would incorporate the results of the staff's evaluation of the technical support documents on defect-specific management which were prepared by the industry. This approach is based on a reduction of the conservatism in the repair criteria being offset by more accurate flaw characterization.

With respect to the other proposals for alternate plugging criteria, all plant specific APC reviews have been suspended pending the generic review of the alternate plugging criteria. A requests from Farley Unit 2 for renewal of its approved interim plugging criteria for a second cycle has been approved and a similar request from Catawba is under review.

Contact:

Jack R. Strosnider, Chief, Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, US Nuclear Regulatory Commission, Washington, DC 20555, (301) 504-2796.

HIGHLIGHTS OF STEAM GENERATOR TUBE PLUGGING CRITERIA

- Steam generator (SG) tubing constitutes a large percentage of the reactor coolant pressure boundary (RCPB) for PWR plants.

- Traditional plugging criteria developed for uniform thinning (i.e., wastage) resulted in a depth-based limit. Defects extending 40% through-wall require repair/plugging.

- The 40% depth-based limit applies to all degradation mechanisms, not just wastage.

- The 40% plugging limit tends to be overconservative for other flaw mechanisms.

- Certain defect-specific plugging limits have been approved in the past, primarily for degradation within the tubesheet (e.g., F* and L* criteria).

- The industry is proposing various forms of alternate plugging criteria (APC) to the traditional depth-based limit for plugging/repair of SG tubes. These include voltage-based and length-based limits.

- A degradation-specific approach to managing steam generator tube integrity has advantages (i.e., it requires using more appropriate inspection and repair criteria for the specific flaw type that is encountered).

- A voltage-based alternate tube repair limit more conservative than that proposed by the industry has been approved (referred to as the interim plugging criteria or IPC) while review of the industry proposal proceeds.

- Significant issues being considered in the staff's review include:

  - Maintaining a low probability of SG tube rupture
  - Evaluating the potential for and consequences of increased leakage under postulated accident conditions

- Implementation of the IPC has actually resulted in more restrictive operating leakage limits.

- A special task force has reviewed the technical bases for the voltage based IPC. The conclusions and recommendations of the Task Group has been issued as a draft report for comment, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes." The public comments are currently under evaluation and will be incorporated in the final report. The staff expects to issue this report as a generic letter.

- The staff is considering a generic approach to defect-specific steam generator inspection and repair criteria. This generic approach would incorporate the results of the staff's evaluation of the technical support documents on defect-specific management prepared which were prepared by the industry. This approach is based on a reduction of the conservatism in the repair criteria being offset by more accurate flaw characterization.

MEDICAL ISSUES

Background:

The Nuclear Regulatory Commission has statutory authority for the regulation of byproduct material as a result of the Atomic Energy Act of 1954. NRC's mandate to protect public health and safety includes the regulation of the medical use of byproduct material in the fields of nuclear medicine, radiation therapy, and research. Consequently, NRC's involvement with, and interest in, this field is not recent, and in fact, the NRC and its predecessor, the Atomic Energy Commission, have regulated the medical use of radioisotopes since 1946. On February 9, 1979, NRC issued a Medical Policy Statement entitled "Regulation of the Medical Uses of Isotopes; Statement of General Policy" (44FR8242). In it, the Commission stated that it will: (1) continue to regulate the medical uses of radioisotopes as necessary to provide for the radiation safety of workers and the general public; (2) regulate the radiation safety of patients where justified by the risk to patients and where voluntary standards, or compliance with these standards, are inadequate; and (3) minimize its intrusion into medical judgments affecting patients, and into other areas traditionally considered to be a part of the practice of medicine.

NRC's oversight of the medical use of byproduct material is focused on ensuring its safe medical use to protect public health and safety. The NRC's emphasis includes protecting workers and visitors from unwanted radiation exposure, as well as assuring that administration of, or radiation from, byproduct material are in accordance with the direction of a physician qualified to use such materials or radiation.

Byproduct material or radiation from byproduct material is regulated by either State or Federal laws. Twenty-nine States, known as Agreement States, have entered into an agreement with the NRC to regulate the use of byproduct material (as authorized by section 274 of the Atomic Energy Act). These States issue licenses and currently regulate approximately 4,500 institutions for medical use, that is, university medical centers, hospitals, clinics, and physicians in private practice. The NRC regulates the medical use of byproduct material in 21 States, the District of Columbia, the Commonwealth of Puerto Rico, and various territories of the United States and administers approximately 2000 licenses for medical use.

Medical Management Plan:

NRC regulation in the area of medical use has prompted criticism and opposition by certain elements within the regulated community. The Commission has been working to effectively resolve these issues and maintain communication with the involved parties. Consequently, during September 1992, a "Medical Issues" paper was developed by the staff. This paper was discussed with the Advisory Committee on the Medical Uses of Isotopes, and representatives from the Agreement States during October 1992, and with NRC regional management in November 1992.

The staff has developed a management plan to include reassessment of the medical use program and initiated a number of actions to address the more pressing problems. Staff initiatives are discussed below.

- Risk Analysis and Human Factors:

  NRC currently has several contracts in place with national laboratories and private entities to evaluate the risks associated with new technologies and the human error component of misadministrations. These include: (1) a contract for the investigation of certain therapy misadministrations, to analyze the root cause(s); (2) contracts to evaluate the contribution of the human-machine interface in brachytherapy and teletherapy performance errors; and (3) contracts to evaluate quality assurance and risk associated with brachytherapy procedures and devices, and gamma stereotactic surgery. These efforts could result in revised equipment and operating procedures, and form the basis for revised regulations and revision of inspection procedures and frequency. Contract work is near completion.

- Medical Use Program Audits:

  As directed by the Commission, independent audits have or will be conducted of NRC's medical use program, in addition to the staff medical management plan. An NRC senior manager conducted a review of the medical use regulatory program and submitted a final report to the Commission during June 1993. The findings of the senior manager's review were considered and incorporated into the staff's proposed medical management plan submitted to the Commission during September 1993. The external review of the program will be conducted by the National Academy of Sciences (NAS). The goal of the external review is to develop an assessment of the adequacy and appropriateness of the current framework for medical use of byproduct material. It will include an in-depth review of the basic regulatory rules, policies, practices, and procedures. The NAS final report is expected in early 1996.

Two key components of the management plan are: 1) continuation of staff initiatives, not only those previously identified to senior management and the Commission, but others that have emerged as the result of recent events; and 2) continuation of interaction with the regulated community. The current initiatives focus on nine major areas of the medical use regulatory program and include such activities as the development of more comprehensive licensing and inspection guidance, and rulemaking. The staff briefed the Commission on the management plan on September 10, 1993, received Commission approval on September 30, 1993, and has begun to implement the plan. The medical management plan is publicly available and is described in SECY-93-244, issued August 31, 1993.

Congressional Hearings:

U.S. Senate Committee on Governmental Affairs
On May 6, 1993, the NRC testified before the U.S. Senate Committee on Governmental Affairs on NRC's national program for regulation of the medical use of radioactive material. One area of interest for the Committee was possible options for NRC's regulatory jurisdiction for medical use of radioactive material to ensure a more efficient national program. As a result of this hearing, an NRC Task Force was established to prepare a report for Commission review and submittal to the Senate Committee on this subject. The Task Force briefed the Commission during late July 1993 on various regulatory options. Subsequently, the Task Force received additional guidance from the Commission to expand their efforts and identify the type of data needed by NRC to further evaluate the impact of implementing various options presented. This revised report was submitted to the Commission and Senate Committee during September 1993.

U.S. House of Representatives Subcommittee on Environment, Energy, and Natural Resources
During late July 1993, NRC staff representing the Offices of the Nuclear Material Safety and Safeguards, and State Programs testified before the U.S. House of Representatives Subcommittee on Environment, Energy, and Natural Resources regarding NRC's oversight of the national medical use regulatory program. In particular, NRC's oversight of Agreement State medical use regulatory programs and their compatibility with NRC's regulatory program were the focus. It is expected that the staff will testify at a potential hearing on other aspects of the NRC's materials program in early 1994.

Update on ACNP/SNM Petition for Rulemaking:

On June 17, 1993, NRC published a proposed rule (58FR33396) to amend its regulations for the medical use of byproduct material. The proposed rule provides greater flexibility by allowing properly qualified nuclear pharmacists and authorized users, who are physicians, greater discretion to prepare radioactive drugs containing byproduct material for medical use. The proposed rule would also allow research involving human subjects using byproduct material and the medical use of radiolabeled biologics.

The comment period expired October 15, 1993, and the staff is scheduled to submit the final rule to the Commission for approval in January 1994.

Additionally, NRC published a final rule on July 22, 1993, (58FR39130) to extend the expiration date of the interim final rule related to preparation and therapeutic use of radiopharmaceuticals from August 23, 1993, to December 31, 1994. This action allows licensees to continue to use byproduct material until NRC completes the related rulemaking described above.

Contact:
Larry Camper, Section Leader, Office of Nuclear Material Safety and Safeguards, US Nuclear Regulatory Commission, Washington, DC 20555 (301) 504-3417

HIGHLIGHTS ON RE-EVALUATION INITIATIVES

- Staff medical management plan was forwarded to the Commission during September 1993. The management plan contains: 1) action items resulting from the findings of the Incident Investigation Team initiated to investigate the patient fatality in November 1992, in Indiana, Pennsylvania; 2) items identified in the senior manager review; and 3) items previously identified by the staff and discussed in the staff medical issues paper developed during September 1992. The staff briefed the Commission on this subject on September 10, 1993, received Commission approval on September 30, 1993, and has implemented the plan.

- Several contracts with national laboratories and private entities involve evaluating risks associated with new technologies and the human error component of misadministrations. Final reports will be submitted within the next two months.

- An internal review by an NRC senior management representative was performed and a report submitted to the Commission during June 1993. An external review will be conducted by the National Academy of Sciences with the final report submitted in early 1996.

- As a result of the May 6, 1993, hearings before the U.S. Senate Committee on Governmental Affairs, an NRC Task Force was established to prepare a report for Commission review that discussed possible options for NRC's regulatory jurisdiction for medical use of radioactive material. The Task Force briefed the Commission during late July 1993 on this subject. Subsequently, the Task Force received additional direction from the Commission to expand their efforts and identify the type of data needed by NRC to evaluate the impact of implementing various options presented. This revised report was submitted to the Commission and U.S. Senate during September 1993.

- During late July 1993, NRC staff representing the Offices of Nuclear Material Safety and Safeguards, and State Programs testified before the U.S. House of Representatives Subcommittee on Environment, Energy, and Natural Resources regarding NRC's oversight of the national medical use regulatory program. In particular, NRC's oversight of Agreement State medical use regulatory programs, and the compatibility of these programs with the NRC's regulatory program were the focus. It is expected that the staff will testify at a potentia hearing on other aspects of the NRC's materials program in early 1994.

INTEGRITY OF REACTOR VESSEL INTERNALS

Background:

Many boiling water reactor (BWR) vessel internals are made of materials susceptible to intergranular stress corrosion cracking (IGSCC), including stainless steel, alloy 600, alloy X750, and alloy 182 weld metal. IGSCC is a time dependent material degradation process, and is known to be accelerated by the presence of crevices, residual stresses, material sensitization, irradiation, cold work and corrosive environments.

Cracking of BWR core shrouds and jet pump hold-down beams have been the most significant of BWR internals cracking reported in 1993. As operating BWRs begin to age, the number of cracking incidents is expected to increase. In anticipation of this trend, the industry is developing a proactive program to monitor and control further cracking of the reactor internals. The General Electric Company (GE) has recommended that BWR utilities perform inspection of both high and low carbon stainless steel shrouds and the Boiling Water Reactor Owner's Group (BWROG) has developed a plan to address the issue. GE is also in the process of developing updated recommendations to address jet pump hold-down beam cracking.

Discussion of Technical Issues:

Core Shroud Cracking:

Cracking of the core shroud was visually observed in 1991 in an overseas BWR. The core shroud is a stainless steel cylinder which separates feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core. The crack in the overseas BWR was located in the heat affected zone of a circumferential weld in the lower shroud. General Electric Company (GE) reported the cracking found in the overseas reactor in Rapid Information Communication Services Information Letter (RICSIL) 054. A number of domestic BWR licensees have recently performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic experience. Core shroud cracking was reported at Brunswick Unit 1 and at Peach Bottom Unit 3 in 1993. Both plants have experienced axial and circumferential cracking of shroud welds located at the core midplane level, and circumferential cracking at the horizontal weld which fuses the lower shroud to the top guide support ring. Some cracking at welds associated with the upper shroud was also found at Brunswick Unit 1.

The cracking at the top guide support ring weld (H-3 weld) at Brunswick Unit 1 is of significant size, with a length extending nearly 360° around the shroud and a maximum depth of at least 1.7 inches deep. Other circumferential and axial cracks were also discovered in the heat affected zones of welds associated with the upper shroud (H-1 and H-2 welds) and welds associated with the mid-shroud (H-4, H-5 and H-6 welds); however, the structural evaluations of these cracks indicate that they are of lesser safety significance than the crack associated with the H-3 weld. Carolina Power and Light Company (CP&L, the licensee for Brunswick Unit 1) proposed a modification of the Brunswick Unit 1 shroud which involves installing a number of mechanical clamps around the degraded H-3 weld. The proposed modification is designed to maintain structural integrity around the H-3 weld for the remaining life of the unit.

The Philadelphia Electric Company (PECo, the licensee for Peach Bottom Unit 3) performed a flaw evaluation and has determined that the cracking in the core shroud is not significant enough to threaten the structural integrity of the shroud during the next operating cycle. The staff agreed with PECo's conclusion that modification of the Peach Bottom Unit 3 shroud is not necessary at this time.

GE currently recommends (Safety Information Letter 572, Rev. 1) that BWR licensees perform visual or ultrasonic inspections of their core shrouds after six cumulative years of power operation if the shroud is fabricated from normal carbon content (0.03% to 0.08% C) austenitic stainless steel, or visually or ultrasonically inspect their shrouds after eight full power years of operation if the shroud is fabricated from low carbon content (< 0.03% C) austenitic stainless steel. GE also recommends that shrouds be reinspected at every subsequent refueling outage or every two refueling outages, depending on whether or not cracking is observed in the shroud.

Jet Pump Hold-down Beam Cracking:

Recently, a jet pump hold-down beam failed by IGSCC at Grand Gulf Unit 1 after about 9 years of service. Two additional hold-down beams at Grand Gulf had ultrasonic indications. Numerous jet pump failures had occurred in the late 1970's and early 1980's. An NRC Bulletin (IEB-80-07, "BWR Jet Pump Assembly Failure") was issued that requested that BWR-3 and BWR-4 licensees perform operability surveillances on the jet pump assemblies. The failure at Grand Gulf was different from prior failures because it occurred in a different location. GE examined the failure and concluded that once a crack begins to grow, it can propagate to failure in less than one operating cycle. Following GE recommendations, Grand Gulf replaced all of the jet pump hold-down beams.

GE currently recommends that licensees with BWR 4, 5, or 6 design reactors replace their jet pump hold-down beams after 8 years of cumulative power operation. The Pennsylvania Power and Light Company (PP&L, the licensee for the Susquehanna Station) has determined that Susquehanna Unit 1 has the same type of hold-down beams as the licensee at Grand Gulf and will have exceeded 9 years of operation prior to the next refueling outage. All jet pump hold-down beams inside the Susquehanna Unit 1 reactor are currently being replaced by PP&L in accordance with the GE recommendations. PP&L has opted for the replacement of the jet pump hold-down beams because GE cannot ensure jet pump hold-down beam structural integrity during the next operating cycle.

Current Status:

GE has developed a set of screening criteria for evaluating the structural integrity of the core shrouds. The GE screening criteria are based on both limit load and linear elastic fracture mechanics methods. GE has also developed a generic safety assessment (GE white paper) of core shroud cracking. The generic assessment evaluates the effects of normal operating, design basis accident and seismic event conditions on postulated 360 circumferential shroud cracks.

The Boiling Water Reactor Owners Group (BWROG) is developing a Core Shroud Cracking Action Plan. The BWROG worked in conjunction with GE to develop the generic safety assessment (GE white paper) and generic core shroud inspection guidance and acceptance criteria for core shroud cracking. The generic safety assessment has been submitted to the NRC for review. In addition, the Action Plan includes plans for compiling and evaluating the data provided by BWR licensees who have performed shroud inspections during 1993 Fall/Winter refueling outages.

The BWROG is also in the process of formulating a jet pump hold-down beam action plan that will be discussed with the staff early in January, 1994. The BWROG has requested that GE provide the BWROG with a listing that indicates the actual operating hours for each GE 4, 5, and 6, the heat treatment (original or modified) for the hold-down beams, and the design for the beams (original or improved), to assist the Owners Group in developing its jet pump hold-down beam action plan.

The staff is keeping close contact with the BWROG and the industry regarding the status of their on-going programs. The staff has met with the BWROG and GE every year since 1988 to review the generic safety implications of potentially IGSCC-susceptible internals and the status of their programs. The staff is in the process of reviewing the BWROG/GE generic safety assessment of core shroud cracking, and will review the generic core shroud inspection guidance and acceptance criteria upon their submittal to the agency. The staff is scheduled to meet with the BWROG in January 1994 to discuss the status of the Owners Group's Action Plan for addressing core shroud cracking and the issue of jet pump hold-down beam cracking in BWRs. The staff will continue to monitor industry actions to address BWR reactor internals cracking.

The staff has issued Information Notice 93-17, "Core Shroud Cracking at Beltline Region Welds in Boiling Water Reactors," to inform the industry of the cracking discovered at the Brunswick Unit 1 reactor. An information notice on jet pump hold-down beam cracking will be issued shortly. The need for further generic communications will be determined as additional core shroud and jet pump hold-down beam inspection data become available from the industry.

The staff has also prepared Preliminary Safety Assessments of jet pump hold-down beam and core shroud cracking. The Safety Assessments discuss the potential consequences of a jet pump hold-down beam failure or a shroud failure during normal, transient, and accident conditions on reactor safety. Based on the assessments, the staff has concluded that there is not an immediate safety concern regarding the BWR core shroud cracking or jet pump hold-down beam cracking, and that there is time for the industry to develop and implement logical programs to address these issues.

Contact:

Jack R. Strosnider, Chief, Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, USNRC, Washington, DC 20555, (301) 504-2796.

HIGHLIGHTS OF BWR INTERNALS CRACKING ISSUES

CORE SHROUD CRACKING

- Cracking was discovered in the core shrouds of the Brunswick Unit 1 and Peach Bottom Unit 3 reactors in 1993. One of the cracks in the Brunswick Unit 1 shroud was a 360° circumferential crack of the weld which joins the top guide support ring to the mid-shroud shell (H-3 weld).

- The staff's current position is that 360° circumferential cracks of any significant depth in the core shroud are unacceptable for continued operation without performing some acceptable method of repair of the cracks - or replacement of the component prior to restart.

- The Carolina Power and Light Company, the licensee for the Brunswick Station, has decided to repair the 360° circumferential crack at the H-3 weld of the Brunswick Unit 1 core shroud by installing 12 mechanical clamps around the degraded weld. The clamps are designed to maintain the structural integrity of the Brunswick Unit 1 shroud at the H-2 and H-3 weld elevations in lieu of the welds themselves.

JET PUMP HOLD-DOWN BEAM CRACKING

- The Entergy Corporation, the licensee for the Grand Gulf Station, discovered a disassembled jet pump assembly inside the Grand Gulf Unit 1 reactor vessel in 1993. After discussions with GE, the licensee decided to replace the jet pump hold-down beams for all jet pump assemblies inside the reactor vessel.

- GE contacted Pennsylvania Power and Light Company, the licensee for the Susquehanna Station and recommended that all the jet pump hold-down beams be replaced. The licensee followed GE's recommendations.

- Based on the assessments, the staff has concluded that there is not an immediate safety concern regarding the BWR core shroud cracking or jet pump hold-down beam cracking, and that there is time for the industry to develop and implement logical programs to address these issues.