INDEX OF BRIEFING PAPERS

BP17 Inspection of Nuclear Power Plants
[NRR]

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]

LICENSE RENEWAL

Background:

Nuclear power provides approximately 20 percent of the electric power produced in the United States. There are 112 licensed nuclear power plants. The term of the initial operating license is limited to 40 years. The first 40 year operating license will expire in the year 2000 and 37 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.

Twenty-eight plants will have their licenses expire by the year 2010. (Note that the 37 percent estimate and 28 plant values does not include potential license extension as a result of construction recapture. Refer to "Information Digest," NUREG-1350, 1992Edition for additional information.) (list attached)

License Renewal:

The Atomic Energy Act limits commercial power reactor licenses to 40 years but also permits the renewal of such licenses. Two rules (10 CFR Part 54 and Part 51) and supporting regulatory guidance establish the technical and procedural steps and further, establish criteria to be used to determine if regulatory requirements are met for license renewal.

As issued, these regulations are based on two key principles. The first principle is that currently licensed facilities are safe. In fact, an average plant invests considerable resources in modifications and updating after initial licensing. Therefore, with the exception of age-related degradation unique to the extended period of operation, the regulatory process provides reasonable assurance that the licensing basis of all currently operating plants provides and maintains an acceptable level of safety for operation. The second principle is that each plant will remain safe during the renewal period. Therefore, each plant's licensing basis must be maintained during the renewal period, in part through a program of age-related degradation management for plant systems, structures, and components.

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 degradation that would occur in the period of extended operation.

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 our 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. Moreover, the Commission is developing further regulatory guidance (Regulatory Guides) and a Standard Review Plan to assure the quality and uniformity of staff review of each submittal.

The Commission's regulations governing license renewal are contained in 10 CFR 54 and require the applicant to describe and justify how they will identify and screen all the structures, systems, 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 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 age-related degradation.

The Commission is working with the industry (under the auspices of the Nuclear Utilities Management and Resources Council) to review a series of technical reports on evaluation of age-related degradation effects of a variety of structures and components important to license renewal. These reports will assist licensees in identifying and screening SSCs.

A parallel rulemaking effort concerning environmental issues related to license renewal is presently in progress. A Generic Environmental Impact Statement (GEIS) has been published and a current modification of 10 CFR 51, environmental impact considerations, has been published for public comment. The public comment period ended in March 1992. This rulemaking is

based on the belief that certain environmental issues should be treated generically as part of this rulemaking.
Such issues would not have to be addressed in a plant-specific environmental report for license renewal.

Current Status:

The Commission expects the initial license renewal review process to take approximately five years based on a detailed technical review and hearing process. The Commission estimates that an applicant would need approximately three to five years to prepare its application. An applicant may apply as early as 20 years before the expiration of its current license.

Two plants were selected by the industry (DOE and EPRI) as lead plants: Monticello in Minnesota and Yankee Rowe in Massachusetts.

The Yankee Rowe plant was shut down in 1991 as a result of questions related to the reactor pressure vessel. In February 1992, the licensee announced its decision not to restart the Yankee Rowe facility and that the plant would be decommissioned. A license renewal application for the Monticello Plant is expected in early 1993.

Contact:
John Craig, Chief, License Renewal Project Directorate
US 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.
- New 10 CFR Part 54 allows for renewal up to 20 years.
- 10 CFR Part 54 provides procedures and requirements for license renewal applications.
- Application review focused on age-related degradation of structure, systems, and components important to license renewal.
- Topical (Industry) Reports being developed which analyze age-related degradation of certain SSCs.
- 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 five years prior to expiration of current license.
- Current issues of note: - Monticello's application scheduled for early 1993.
- Yankee Rowe's decision to decommission, announced February 1992.

PLANTS WHOSE LICENSE EXPIRES BEFORE 2011*

Expiration Date	Plant Name	Plant Type	Generating Capacity (MWe)	NRC Region
2000	Yankee Rowe	PWR	167	I
	Big Rock Point	BWR	67	III
2004	Oyster Creek 1	BWR	620	I
	San Onofre 1	PWR	436	V
2005	Nine Mile Point 1	BWR	615	I
2006	Dresden 2	BWR	772	III
2007	Haddam Neck	PWR	565	I
	Palisades	PWR	730	III
	Turkey Point 3	PWR	666	II
	Turkey Point 4	PWR	666	II
2008	Diablo Canyon 1	PWR	1073	V
	Fort Calhoun	PWR	478	IV
	Maine Yankee	PWR	830	I
	Peach Bottom 2	BWR	1055	I
	Peach Bottom 3	BWR	1035	I
	Zion 1	PWR	1040	III
	Zion 2	PWR	1040	III
2009	D.C. Cook 1	PWR	1020	III
	D.C. Cook 2	PWR	1060	III
	Ginna	PWR	470	I
	Indian Point 3	PWR	965	I
2010	Brunswick 1	BWR	790	II
	Brunswick 2	BWR	790	II
	Diablo Canyon 2	PWR	1087	V
	H.B. Robinson 2	PWR	665	II
	Millstone 1	BWR	654	I
	Monticello	BWR	536	III
	Point Beach 1	PWR	485	III

DESIGN CERTIFICATION PROCESS

Background:

The Commission has long sought nuclear power plant standardization to enhance safety and bring stability and predictability to licensing. Standardization has the double aim of enhancing safety and making it possible to resolve design issues before construction starts. Pre-construction resolution of design issues could be achieved through combined construction permit and operating license reviews. The Commission has also sought to improve the licensing environment for future nuclear power reactors by minimizing the complexity and uncertainty in the regulatory process. The Commission approved a new rule, Part 52, on April 18, 1989, which sets out a 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 long sought goal of enhanced safety and early resolution of licensing issues.

Design Certification:

The Division of Advanced Reactors, responsible for design certification, was created in the fall of 1991, 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 Part 52. Generally, the staff conducts its reviews of advanced reactor plant designs in accordance with Commission guidance of the Severe Accident Policy Statement (50 FR 32138; August 8, 1985), Standardization Policy Statement (52 FR 34884; September 15, 1987), and Safety Goal Policy Statement (51 FR 30028; August 21, 1986). These reviews are expected to result in the certification of individual designs at the completion of individual design certification rulemaking.

Design certifications are issued for a duration of fifteen years permitting operating experience with a given design to accumulate before the certification comes up for renewal.

Regulations:

The Commission's Part 52 requires the applicant(s) to provide the technical information necessary to demonstrate compliance with the technically relevant standards set out for construction permits and operating licenses in Parts 50, 20, 73, and 100. Generally, Part 52 sets out the requirements and provisions applicable to situations in which applications are filed by one

or more applicants for licenses to construct and operate nuclear power reactors of essentially the same design to be located at different sites. Subpart B of 10 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 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 Part 52 relates to the involvement of the public in the design certification process. Part 52 streamlines public participation by moving the bulk of the issues up front in the licensing process, to the design certification part of the process.

Under 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, a design-specific probabilistic risk assessment, inspections, tests, analyses, and acceptance criteria (ITAAC), interface requirements, and ITAAC for the interface requirements. Those portions of the design that are either site specific or include structures, systems, and components which do not affect the safe operation of the facility may be excluded from the scope of the design.

Current Status:

The status of advanced light water reactor reviews was most recently updated in a briefing to the Commission on October 29, 1991. Specifically:

Electric Power Research Institute (EPRI) Requirements Document: The staff expects to issue the Final Safety Evaluation Report (FSER) on the EPRI Evolutionary Requirements in August 1992. The Draft Safety Evaluation Report (DSER) on the EPRI Passive Requirements is scheduled for April 1992, followed by the FSER in September 1993.

General Electric Advanced Boiling Water Reactor (ABWR): Completion of the Safety Evaluation Report is scheduled for December 1992. Issuance of the ABWR design certification is scheduled for June 1994.

Combustion Engineering System 80+: Application essentially completed on March 4, 1991; entering the Draft Safety Evaluation

Report stage. Completion of the Safety Evaluation Report is scheduled for November 1993, with design certification scheduled for May 1995.

Westinghouse AP600: Staff expects AP600 application for final design approval in June 1992. Completion of the Safety Evaluation Report is scheduled for November 1994.

General Electric Small Boiling Water Reactor: Application due in August 1992. Completion of the Safety Evaluation Report is scheduled for January 1995.

Contact:

Robert C. Pierson, Chief, Standardization Project Directorate
US NRC, Washington, DC 20555 (301) 504-1118

HIGHLIGHTS OF DESIGN CERTIFICATION PROCESS

o The Commission has long sought nuclear power plant standardization.

o Standardization will enhance safety.

o Licensing environment improved by minimizing complexity and uncertainty.

o Design certification is the key procedural device for bringing about enhanced safety.

o Division of Advanced Reactors, responsible for design certification established in the fall of 1991.

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

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

o 10 CFR 52 streamlines public participation by moving the bulk of the issues up front to the design certification stage.

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

o Commission meeting on Status of Advanced Reactor Reviews - October 29,1991.

o EPRI Requirements Document review is nearing completion.
Evolutionary FSER scheduled for August, 1992.
Passive DSER scheduled for April, 1992.

o General Electric ABWR FSER is scheduled for December, 1992.

o Application for Combustion Engineering System 80+ design essentially complete on 3/4/91; entering Draft Safety Evaluation Report stage.

o Approval of application for Westinghouse AP600 certified design expected in June, 1992.

BELOW REGULATORY CONCERN POLICY

Background:

On July 3, 1990, the U.S. Nuclear Regulatory Commission (NRC) published in the Federal Register its Below Regulatory Concern (BRC) Policy Statement. This policy established a framework to guide the Commission in making future decisions on whether to grant exemptions from Commission regulations in cases where radiation levels are so low that they do not require the imposition of stringent regulatory controls to ensure protection of the public health and safety. The policy statement was the culmination of a number of Commission efforts to address BRC concerns.

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 BRC Policy from taking effect.

Current Status:

In response to public concerns, the Commission declared a moratorium on the implementation of the BRC policy on June 28, 1991, in order to initiate a consensus-building process on BRC issues. Although the consensus process received widespread and strong support from state government, the nuclear industry, the medical community and others, representatives of national environmental interests ultimately declined to participate in the process. Based on the lack of participation by a major interest affected by the BRC policy, the Commission decided to discontinue its effort to build consensus. The Commission is continuing its moratorium on implementation of the BRC policy.

One of the more controversial aspects of the BRC policy was the issue of Agreement State compatibility. The Commission intends to take a fresh look at this issue. It is interested in the advantages and disadvantages of a national approach on radiation safety matters, and is also trying to determine a mechanism to provide the Agreements States with flexibility to address local needs and conditions. On December 23, 1991, the Commission published a Federal Register notice asking for comments that will provide the basis for establishing a Commission policy. The Commission is soliciting recommendations from the full spectrum of interested parties on how to implement a policy on compatibility of Agreement States policy.

Finally, as an alternate to the consensus process, the Commission requested, and the staff has prepared a plan for proceeding with a participative rulemaking that would provide early access to affected interests on the development of a proposed rule on the radiological criteria for decommissioning and decontaminating of affected sites with residual radioactivity. The Commission is considering the staff's recommendations on the participative rulemaking.

Contacts:

Francis X. Cameron, OGC (301) 504-1625
Donald A. Cool, RES (301) 492-3785
Cynthia G. Jones, NMSS (301)504-2629

HIGHLIGHTS

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

- o The BRC Policy Statement was issued on July 3, 1990 in the Federal Register.
- o In response to public concern, the Commission declared a moratorium on the implementation of the BRC policy on June 28, 1991.
- o With this moratorium, NRC initiated a phased consensus-building process on BRC issues.
- o On November 12, 1991, the last of the potential representatives of environmental interests informed NRC that it would be unable to represent a major interest in the BRC consensus process.
- o At this time, due to the lack of participation in the environmental community, the Commission has decided to abandon its effort to build consensus.
- o As an alternate to the consensus process, the Commission is considering a plan for proceeding with an enhanced rulemaking that would provide early access to affected interests on the development of radiological criteria for decommissioning.

- With respect to the Agreement State compatibility issue, the Commission published a *Federal* Register notice on December 23, 1991, asking for comments that will provide the basis for establishing a Commission policy on this issue.

- The Commission is continuing its moratorium on implementation of the BRC policy.

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 office's 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 physic 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 promulgations 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 1991, there were 112 operating nuclear power plants. For these plants, there were 172 resident inspectors and 255 regional office inspectors.

Contact:

Mark W. Peranich, Chief, Inspection Program Development Section,
Inspection and Licensing Program Branch
US NRC, Washington, DC 20855. (301) 504-3078

HIGHLIGHTS OF RESIDENT INSPECTOR PROGRAM

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

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

- o 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.

- o In 1977, the NRC initiated the resident inspection program.

- o 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.

- o As of 1991, there were 112 operating nuclear power plants. For these plants, there were 172 resident inspectors and 255 regional office inspectors.

YANKEE ROWE DECOMMISSIONING DECISION

Background:

The reactor pressure vessel is a principal component in the reactor coolant system of a light water cooled reactor. The reactor pressure vessel forms part of the pressure boundary for the cooling water and the structural support for the reactor core. When the nuclear fission process is taking place in the reactor core, high energy neutrons are produced. Some of these neutrons interact with the uranium fuel to sustain the fission chain reaction. Others are absorbed by materials specifically designed to absorb the excess neutrons, such as control rods or soluble boron poison added to the coolant, while others may exit from the core region. A portion of the neutrons that exit from the core region strike the reactor pressure vessel wall. This neutron radiation that bombards the vessel can displace some of the atoms in the crystal structure of the metal in the vessel wall, reducing the toughness of the material. This phenomenon is called neutron embrittlement because the metal becomes more brittle and less flaw tolerant.

The Nuclear Regulatory Commission(NRC) has develcped regulations which are designed to ensure that the integrity of the reactor vessel is maintained under all normal operating and potential accident conditions. In particular, these are 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements;" 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements;" 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events;" and 10 CFR 50.55a, "Codes and Standards."

During the NRC staff's review of the Yankee Rowe reactor vessel, concerns were raised over the degree of embrittlement of the vessel and specifically, whether the vessel had adequate margin against failure in the event of a postulated pressurized thermal shock (PTS) event. PTS events are transients that can possibly occur in a pressurized water reactor and are characterized by severe overcooling, concurrent with, or followed by, repressurization of the coolant. To assess the vessel's ability to withstand the PTS event, the NRC staff has developed PTS screening criteria. Should the vessel properties reach these criteria, the licensee is required to provide detailed fracture mechanics analyses to show that sufficient margin remains against failure.

Regulatory Actions:

As a result of discussions between the staff and the licensee (Yankee Atomic Electric Company) in April and May 1990, the

licensee submitted its analysis of the Yankee Rowe vessel. In this analysis, dated July 5, 1990, the licensee concluded that the screening criteria would not be reached until 2020. However, the staff believed that the Yankee Rowe reactor vessel may have already reached the screening criteria. The staff's concerns were, in part, based on the limited information that was available regarding the actual chemical and physical characteristics on the vessel to support the assessments. In a letter dated August 31, 1990, the staff provided the results of its review of the licensee's probabilistic vessel failure analysis, concluding that substantial uncertainties associated with the weld chemistry and the toughness characteristics of the plate materials existed. However, the staff judged, after balancing these uncertainties against the low probability of a PTS event occurring, that it was acceptable for Yankee Rowe to be operated until the end of its present fuel cycle. The licensee proposed to address the uncertainties through a plan which included: (a) inservice inspection of the beltline materials, (b) weld sample removal and chemical analysis, and (c) an accelerated irradiation test program to determine the effect of irradiation on the plate materials.

On June 4, 1991, the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution petitioned the Commission for emergency enforcement action against Yankee Rowe. The petitioners asked that the plant be shut down because it "is operating in violation of NRC requirements for pressure vessel integrity." On June 25, 1991, the Director of the Office of Nuclear Reactor Regulation denied the petitioners' request for emergency relief and indicated that, as required by 10 CFR 2.206, the NRC would address the specific issues raised.

On July 11, 1991, the staff, the petitioners, and the licensee briefed the Commission on the petition. The Commission indicated that it would make the final determination on whether or not the Yankee Rowe plant would be required to shut down as requested by the petitioners.

On July 24, 1991, the staff submitted SECY 91-220 to provide the Commission with the proposed Director's Decision and other relevant information about the Yankee Rowe pressure vessel embrittlement issues.

On July 29, 1991, the Commission was briefed again by the parties and on July 31, 1991, met and affirmed the issuance of Memorandum and Order CLI 91-11 which concluded that continued operation of the Yankee Rowe Plant for an interim period will not pose undue risk to the public health and safety. In CLI-91-11, the Commission ordered the licensee to investigate certain measures that could further reduce the probability of vessel failure and submit a plan to resolve uncertainties in the chemical and metallurgical characteristics of the reactor vessel.

Furthermore, the licensee was required to report the results of its efforts in response to the Commission order by August 26, 1991, and on a monthly basis thereafter.

On September 30, 1991, the NRC staff issued a memorandum to the Commission recommending that Yankee Rowe be shut down until the NRC was satisfied that the reactor pressure vessel has adequate margin against vessel failure from a PTS event. The memorandum addressed differences between the licensee's analysis submitted on August 26, 1991, and the staff's review of that submittal, as supplemented. In response to the memorandum, the licensee elected to voluntarily shut down the plant on October 1, 1991. On October 2, 1991, the staff issued a Confirmatory Action Letter confirming the licensee's commitment to obtain NRC approval prior to restart of the plant.

On December 19, 1991, the licensee presented to NRC management and staff the details of its initial restart assessment and proposed restart criteria. This information, presented in a letter dated December 18, 1991, responded to the actions requested in the letter of October 30, 1991, from the Director, Office of Nuclear Reactor Regulation. These actions included providing an overall plan and criteria which focused on the long-term restart of the facility following the licensee's decision not to seek a restart prior to the 1992 outage. During January and early February 1992, the staff and licensee met on numerous occasions for detailed discussions on the various aspects of the licensee's restart action plan. Based on the plan and the information received from these interfaces, the staff provided its proposed restart criteria to the licensee in a letter dated February 14, 1992.

On February 27, 1992, the licensee notified the Commission that, effective immediately, it would permanently cease power operation of Yankee Rowe and would begin developing plans to decommission the facility in accordance with 10 CFR Part 50. The decision was based on a combination of factors, most importantly the economic outlook and the degree of uncertainty associated with resolution of the reactor vessel issues raised by the NRC staff.

Current Status:

The licensee is preparing a request for a possession-only license (POL) to cover the storage of the nuclear fuel on-site in the spent fuel pool. Additionally, work is beginning on the preparation of amendments to the facility technical specifications related to the possession-only status. A detailed

decommissioning plan required by 10 CFR 50.82 will be submitted within two years of February 27, 1991, the date that the licensee notified the Commission of its decision to permanently cease power operation of Yankee Rowe.

Contact:
Steven A. Varga, Director, Division of Reactor Projects
US NRC, Washington, DC 20555 (301) 504-1403

YANKEE ROWE (YR) DECOMMISSIONING DECISION

- o RPV is a pressure boundary that provides a fission product barrier.
- o RPV must have sufficient fracture toughness to resist fracture during a pressurized thermal shock event - 10 CFR 50.61.
- o April-May 1990, the staff explained to the licensee its concern related to fracture toughness of the YR RPV.
- o On July 5, 1990, the licensee provided an analysis of the YR RPV.
- o On August 31, 1990, the staff concluded that: (a) uncertainties in material properties could result in the YR RPV exceeding the screening criteria in 10 CFR 50.61, (b) YR was safe to operate to the end of fuel cycle 21, and (c) the licensee should implement a plan to reduce uncertainties.
- o On July 31, 1991, the Commission issued a Memorandum and Order which sought a reduction in probability of vessel failure by a factor of 5 to 10.
- o On August 26, 1991, the licensee submitted the results of its re-analysis to the Commission.
- o On September 30, 1991, the staff recommended that YR be shutdown because the results from the new thermal-hydraulics model substantially reduced the staff's confidence that the calculations of the likelihood of vessel failure from PTS events are conservative and bounding values were used throughout the analysis.
- o On October 1, 1991, the licensee began a safe and orderly shutdown of YR.
- o On December 19, 1991 the licensee presented its restart criteria to the NRC. Based on its review the staff provided its proposed restart criteria on February 14, 1992.
- o On February 27, 1992 the licensee notified the Commission of its intent to cease opoeration and plan for decommissioning. The decision was based on several factors, most importantly economic and uncertainty with the resolution of the reactor vessel technical issues.

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; (2) recognizes that equipment can fail and operators can make mistakes, therefore requires 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 requires 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., 112 commercial nuclear power reactors are currently licensed to operate at 70 sites. For each reactor site 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 annually. They also exercise biennially State and local government emergency plans for each operating reactor site with participation of State and local governments within the plume exposure EPZ.

Contact:

Cortact:
Robert Erickson, Chief, Emergency Preparedness Branch
U.S. NRC, 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 emergency 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.
- Nucleaı power reactor licensees exercise their emergency plans annually.
- Licensees exercise their emergency plans with those of offsite authorities within the plume EPZ 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.

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

Emergency Response Data System (ERDS):

ERDS will provide 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 requires each licensee 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 may also be 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, and Ohio have signed an ERDS MOU and eight (8) 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. 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 are required to implement ERDS prior to February 13, 1993. Licensee implementation plans, due in the near future, will include projected utility schedules for completing the ERDS implementation.

Contact:

Gary G. Zech, 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.

- Twenty-two licensee reactor units have implemented ERDS under a voluntary program.

- All other 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.)

- ERDS supplements the currently installed ENS.

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

Background:

The Systematic Assessment of Licensee Performance (SALP) program was established by the NRC following the Three Mile Island accident. At the time of the accident, the NRC relied on periodic NRC inspections to identify problems with plant performance. With the institution of the SALP process, the NRC developed a routine, systematic approach for the review of all inspection and licensing activity. The SALP process is used by the NRC to synthesize its observations of and insights into a licensee's performance and to identify common themes or symptoms. As such, the NRC needs to recognize and understand the reasons for a licensee's strengths as well as weaknesses. The primary product of the SALP process is the SALP report. The results of the SALP process, as documented in the SALP report, are used to express NRC senior management's observations and judgements on licensee performance. The SALP process recognizes both strengths and weaknesses in a facility. The SALP process is not intended to identify proposed resolutions or solutions to problems. The licensee's management is responsible for ensuring plant safety and establishing effective means to measure, monitor, and evaluate the quality of all aspects of plant design, hardware, and operations.

Program Objectives:

The SALP program has four objectives. The first objective is to improve the regulatory program by focusing NRC management attention on areas of concern. Through their participation in the SALP process, NRC managers identify common themes or symptoms in the licensee's performance. This information is used to determine if changes to NRC requirements are necessary. The second objective of the SALP program is to be instrumental in improving licensee performance. Through the issuance of a SALP report, the licensees have a clear understanding of the NRC concerns and can act to correct the conditions the NRC identifies as significant. The third objective of the SALP program is to provide a mechanism that focuses attention on the overall effectiveness of management including underlying strengths and weakness. Even though the proper systems and administrative procedures are in place, plant management must effectively guide, direct, and provide resources for safe plant operations. The fourth objective of the SALP program is to assist the NRC in the allocation of inspection resources. Plants with good performance history are considered for reduced inspection effort, while plants with poor performance history are considered for increased inspection effort.

SALP Program Requirements:

The SALP program assesses licensee performance in seven functional areas. These functional areas include plant operations, radiation protection, maintenance/surveillance emergency preparedness, security, engineering/technical support, and safety assessment/quality verification. Additional functional areas may be added if the NRC staff believes it is necessary.

The seven functional areas are evaluated against six criteria. The six criteria are (1) the assurance of quality, including management involvement and control, (2) approach to the resolution of technical issues from a safety standpoint, (3) enforcement history, (4) operational events, (5) staffing, and (6) effectiveness of the training and qualification program.

A performance rating is assigned in one of three categories for each of the above functional areas. These performance categories are defined as follows:

Category 1	Licensee management attention to and involvement in nuclear safety or safeguards activities resulted in a superior level of performance. NRC will consider reduced levels of inspection effort.
Category 2	Licensee management attention to and involvement in nuclear safety or safeguards activities resulted in a good level of performance. NRC will consider maintaining normal levels of inspection effort.
Category 3	Licensee management attention to and involvement in nuclear safety or safeguards activities resulted in an acceptable level of performance; however, because of the NRC's concern that a decrease in performance may approach or reach an unacceptable level, NRC will consider increased levels of inspection activity.

It should be noted that the lowest SALP rating, Category 3, is acceptable performance. 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 15 months. The NRC regional offices can adjust the length of the SALP period plus or minus 3 months 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 is convened in the NRC regional office. The SALP Board membership includes NRC regional and headquarters managers responsible for the oversight of the facility, the NRC senior resident inspector, and the NRC project manager. Board members are selected to represent a diverse group of inspection and license review activities within the NRC. A draft SALP report is prepared by regional and headquarters staff who have had inspection or review responsibilities for the facility. Presentations by those responsible for inspection and review activities at the facility are made to the Board. The Board has the opportunity to question those responsible for drafting the report concerning their input. The Board reviews the draft SALP report in detail and discusses plant performance in relationship to the SALP criteria. At the conclusion of the discussion of each functional area, the Board assigns a category rating and can make recommendations for licensee and NRC action to address the issues in the SALP report.

The Regional Administrator approves the SALP report and sends it to the licensee (and the public document room) as the initial SALP report. The licensee is given the opportunity to review the report. Subsequently, a special public meeting is convened between senior NRC managers and the licensee to discuss the results of the SALP report. This SALP management meeting is normally held near the facility. Following this presentation, the licensee is given the opportunity to comment on the report in writing. When written comments are provided, the NRC will conduct a review of the comments and may change the SALP report based on statements of fact presented in the licensee's written reply. The report is then issued as a final SALP report with an explanation of any changes. If there are no applicable licensee comments, the initial SALP report is considered the final SALP report.

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. Additionally, issues highlighted by the SALP report that may represent a generic issue may be considered for increased inspection. These inspections may result in a staff recommendation to the Commission for changes to the NRC's requirements.

Contact:
E. William Brach, 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 focus NRC management attention on areas of concern, (2) to be instrumental in improving licensee performance, (3) to provide a mechanism that focuses NRC attention on the overall effectiveness of licensee management, and (4) to assist the NRC in the allocation of inspection resources.

- Seven functional areas are evaluated in the SALP process using six criteria.

- A SALP Board comprised of NRC regional and headquarters managers convenes to assess licensee performance.

- The SALP board rates the licensee's performance in three categories, with Category 1 being superior performance and Category 3 being acceptable performance.

- A public meeting is held between the NRC and the licensee to discuss the results of the SALP report.

- A final SALP report is issued following NRC's evaluation of any written licensee comments.

- Good performance (Category 1) can result in longer SALP assessment periods and reduced inspection effort for the licensee, and poor 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 in the 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. 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 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 the possible deterioration of structures and consequent rearrangement of materials inside, which could cause an additional release of radioactive material, mainly as dust. The Soviet authorities are studying options for dealing with the sarcophagus problem and are seeking ideas and help from the international community.

Public concern in the areas near Chernobyl, in the Ukraine, about the safety of the still operating units has led to a Ukrainian government decision to shut them down permanently in 1993.

Within the past two years decisions were taken to evacuate substantial additional nearby areas, in the Ukraine and Byelorussia (and a small area in Russia). About 200,000 people have been relocated so far (including the originally evacuated 135,000). The eventual total is expected to be about 380,000 persons.

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.

International Studies:

Soviet 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 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.

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 are

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, 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 is suspected to be issued in early 1992. That report will close 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.

- 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. Soviet scientists 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 are 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. These are being completed. A report is expected in early 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 licensee's major efforts over the past 12 years have included partial facility decontamination, removal of fuel from the reactor vessel, shipment offsite of substantial quantities of both high and low level radioactive wastes and the removal, treatment and partial 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 by the licensee.

Recent Accomplishments:

The licensee's fuel removal program was completed in April 1990. The reactor vessel, the remainder of the reactor coolant system, the reactor building, and the auxiliary and fuel handling building were defueled to the extent reasonably achievable. The possibility of an inadvertent criticality has been precluded. All canisters containing core debris have been shipped offsite to Idaho National Engineering Laboratory.

During July and August of 1991, the reactor vessel was drained to make final measurements of the residual fuel remaining in the vessel. Less than 1390 pounds (609 kilograms) of fuel have been previously estimated by visual measurements to remain in the reactor vessel. The recent final vessel fuel measurement program is the last phase in a special nuclear materials accountability program at TMI-2. The measurement technique made use of an array of helium filled detectors to measure fast neutrons produced by the residual fuel.

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. The staff will continue to

monitor and evaluate GPUN's reactor vessel fuel measurement program.

Evaporation of the treated accident generated water began in January 1991 after a prolonged period of system testing, modification, and repair. As of October 1, 1991, more than 843,000 gallons, or 37 percent, of the 2.3 million gallons of accident generated water had been decontaminated and vaporized.

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. Through early FY91 the licensee had issued 14 amendments to this SAR. The licensee's request for a possession only license as well as other changes to the license that would allow for PDMS are currently under review by the NRC staff. The staff plans to issue a Safety Evaluation Report (SER) on PDMS early in FY92. The staff's SER and the August 1989 Final Supplement 3 to the Programmatic Environmental Impact Statement for TMI-2 will form the basis for the staff's position on the acceptability of PDMS. On April 25, 1991 the staff published a notice of opportunity for a prior public hearing regarding the licensee's request to amend its license. A member of the public has requested to intervene in the license amendment proceedings.

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 millon in an escrow account for the radiological decontamination of TMI-2. The licensee stated that the accumulation period for the funds would end when the TMI-1 license expires. The $195 million is in excess of the amount required by NRC regulations for a facility the size of TMI-2. The $195 million would likely not cover all costs associated with the remainder of the cleanup and decommissioning. The NRC staff's review of the licensee's plan is ongoing.

Contact:

Michael T. Masnik, Senior Project Manager for TMI-2, Non-Power Reactors, Decommissioning and Environmental Project Directorate US NRC, Washington, DC 20555 (301) 504-1191

HIGHLIGHTS OF THREE MILE ISLAND UNIT 2 STATUS

- Accident occured 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 $195 million for radiological decommissioning.
- Total occupational exposure 6386 person rem as of the end of calen' r year 1990.
- Annual exposure less than 0.1 millirem to the maximally exposed member of the public during defueling.

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 one hundred and eleven commercial nuclear power plants currently in operation in the United States. The Performance Indicator (PI) Program is one of several tools used by senior NRC management in making decisions regarding plant-specific regulatory programs. Performance indicators are intended to provide information concerning nuclear power plant performance trends and to assist NRC management in identifying poor or declining safety performance as well as good or improving safety performance.

Performance Indicator Development:

An interface task group began development of the Performance Indicator (PI) Program in May of 1986. 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 have been provided quarterly to NRC management since then.

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 of programmatic causes as identified in Licensee Event Reports. Each quarter, the AEOD staff provides a report 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 to licensee managers plant-specific information and industry average data extracted from each PI report.

It should be recognized that PIs have limitations and are subject to misinterpretation. The PI Program is a single, coordinated, overall NRC program that provides an additional view of operational performance to enhance NRC's ability to recognize

areas of changing safety performance at 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. Performance indicators alone do not provide a basis for ranking individual power plants or taking regulatory actions and are not used with licensees as an overall measure of plant performance level. The PIs for a given plant, when viewed as a set, provide an additional [illegible] of plant operational performance. They aid in identification of performance and focus attention on assessing and understanding underlying causes (factoring in other available information).

The NRC will continue to review, evaluate, and revise the PI program. Additional indicators will be considered as they are developed.

Contact:

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

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 acceptable 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 reduce this risk to 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, primary 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 is provided for cooling it. Here again, the "defense-in-depth" concept is to provide diverse and multiple backup systems so that there is an adequate assurance of a supply of water for cooling the core. Due consideration is made with respect 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 assure 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 the criteria and requirements for assuring 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 poses no significant additional risk to individuals' life and health and that societal risk is comparable to other viable competing energy supply technologies and there is 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% the total cancer risk from all other causes.

Current Status:

Currently, 112 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 probabilistic risk assessments of their plants (the Individual Plant Examination program). 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 reduce the risk even further. 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. Weaknesses that are found will be addressed through appropriate plant modifications. The second phase will address 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 three years.

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:

William D. Beckner, Chief, Risk Applications 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 unidentified weaknesses that may exist and reduce the risks even further.

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, 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 inserted between and among the pressure tubes. The CANDU 3 design is an evolution of the CANDU 6 design (600 - 640 MWe), which has been approved by the Atomic Energy Control Board (AECB), the Canadian government agency responsible for regulating atomic energy 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, 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 four technology transfer reports describing various aspects of the CANDU technology.

The NRC has tentatively scheduled a Preapplication Safety Evaluation Report (PSER) to be issued in June 1993. In a letter dated September 16, 1991, AECL Technologies stated their intent to file a Standard Design Certification application in the 1995 to 1996 timeframe.

PIUS

PIUS 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. At these two openings, density locks, consisting of a number of open tubes, are provided. During normal operation at design power, the density locks prevent the entry of the highly borated pool water 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, bypassing both the top and bottom density locks. 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. A natural circulation flow path is then established from the borated pool through the lower density lock, up through the core, and back into the borated pool through the upper density lock for long term shutdown cooling. 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 Process Inherent Ultimate Safety (PIUS) Preliminary Safety Information Document (PSID). The preapplication review of the PIUS PSID will continue until the NRC staff issues its final Preapplication Safety Evaluation Report (PSER) to the Commission, tentatively scheduled for July of 1993.

MHTGR

The modular high temperature gas-cooled reactor (MHTGR) design is a helium-cooled and graphite-moderated thermal power reactor. The fuel is composed of millions of ceramic coated microspheres distributed in cylindrical graphite-like rods which are inserted in holes 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. This is about half the output of current light water reactor plants. 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 design MHTGR, 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 now being focused on by the staff are fuel design and performance, containment design and performance, the reactor cavity cooling

system, accident selection and analysis, the selected 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 end of 1992.

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 to update the draft PSER and issue a final PSER is tentatively scheduled for completion by the end of 1992.

Contact:

James E. Dyer, Chief, Advanced Reactor Project Directorate
US NRC, Washington, DC 20555 (301) 504-1111

HIGHLIGHTS - ALL PROJECTS

PRISM --

- o Preliminary Safety Information Document (PSID) submitted to NRC in November 1986

- o NRC Draft Preapplication Safety Evaluation Report (PSER) on PRISM (NUREG-1368) released September 1989

- o DOE submitted Amendments 12 and 13 responding to issues, March and May 1990

- o NRC tentative schedule for final PRISM Preapplication Safety Evaluation Report is November 1992

MHTGR --

- o Preliminary Safety Information Document submitted by DOE in 1986.

- o Draft Preapplication Safety Evaluation Report issued by NRC in March 1989.

- o NRC tentative schedule for final PSER for the MHTGR design is December 1992.

PIUS --

- o ABB requested a licensability review of the PIUS design in October 1989

- o ABB submitted the PIUS PSID for review in May 1990

- o First review of PIUS began in June 1991

- o Final PSER tentatively scheduled to be issued to the Commission July 1993

CANDU 3 --

- o 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.

- o A number of documents were submitted to NRC for preapplication review between mid-1990 and early 1991.

- o 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.
- o The final Preapplication Safety Evaluation Report for the CANDU 3 is tentatively scheduled to be issued in June 1993.
- o In September 1991, AECL Technologies stated their intent to submit the CANDU 3 design for standard design certification in the 1995 to 1996 timeframe.

HIGH-LEVEL RADIOACTIVE WASTE

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. Specifically defined, HLW is primarily in the form of spent fuel discharged from commercial nuclear power plants; it also includes come 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 facility (MRS) by 1998, and a permanent HLW repository deep beneath the surface of the earth by the year 2010. The United States Department of Energy (DOE) has the responsibility for disposing of HLW. The United States Environmental Protection Agency (EPA) is responsible for developing appropriate environmental standards for HLW. The United States Nuclear Regulatory Commission (NRC) has the licensing authority for the disposal and 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 a monitored retrieval storage facility (MRS) in which waste can be stored prior to permanent disposal. NRC has the licensing and related regulatory authority for both the MRS and high-level waste 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. NWPA allows a dual approach to MRS siting: (1) siting by the 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, limited subsurface lateral excavations and borings, and insitu 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 requirements 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 10 CFR Part 72. These requirements 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 requirements for the disposal of HLW in a geologic repository are contained in 40 CFR Part 191. These requirements 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.

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 the 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 and also observes DOE quality assurance audits. All prelicensing consultation activities are open to participation by the State of Nevada, affected Indian Tribes and units of affected local governments.

The 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. However, in

specific areas, the staff identified two objections to DOE starting site characterization. One objection concerns the DOE quality assurance (QA) program, and the other is related to the process for designing the shaft to be used for underground exploratory work. In addition, 196 concerns in the form of comments and questions were raised.

DOE is presently making progress to resolve NRC's site characterization objections and concerns. Regarding quality assurance, DOE has been making considerable progress towards resolution of NRC concerns. There has been partial closure on several particulars involving the QA objection. For example, the NRC staff has concurred with DOE's findings that several DOE contractor programs are acceptable for new site characterization work. With respect to the objection concerning the exploratory studies facility, DOE has conducted a study of alternatives for conducting underground exploration and based on NRC/DOE interactions in 1991, it appears that DOE is adequately considering NRC's concerns. Finally, on the basis of DOE responses, 59 of the 196 other concerns have been closed to date.

The State of Nevada has granted DOE air quality and water injection and appropriations permits, thereby allowing DOE to proceed with surface-based site characterization activities.

The current DOE MRS strategy for meeting the 1998 date (called for in the NWPA) for accepting spent fuel from utilities is to rely on the Nuclear Waste Negotiator for siting an MRS. 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. The following have been awarded Phase I MRS grants to conduct feasibility studies: Mescalero Apache Tribe, New Mexico; Grant County, North Dakota; Fremont County, Wyoming; Yakima Indian Nation, Washington; Sac and Fox Nation and Chickasaw Nation, Oklahoma; and Prairie Island Tribal Council, Minnesota. The Sac and Fox Nation reconsidered and withdrew their application on March 4, 1992; therefore, funds were deobligated. The Mescalero Apache Tribe have gone on to apply for a Phase II grant to continue their feasibility study. A number of other interested States and Indian Tribes have submitted MRS grant applications for review.

EPA developed generally applicable environmental standards in 1985. These standards were remanded in 1987 due to inconsistencies with other standards with respect to individual and ground-water protection. Revised standards are schedule . to be released for public comment in 1992.

<u>Contact:</u>

Joseph J. Holonich, Director, Repository Licensing and Quality Assurance Project Directorate, Office of Nuclear Material Safety and Safeguards, U.S. NRC, Washington, DC 20555. (301) 504-2442

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.
- 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 the NRC an MRS application to construct and operate a facility in 1995.
- DOE to begin waste acceptance at an MRS facility in 1998.
- DOE to submit to the NRC a HLW repository licensing 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.

Cleanup of Radioactivity Contamination at Sites

Background:

There are about 40 sites contaminated with radioactive material under the jurisdiction of the Nuclear Regulatory Commission (NRC) throughout the country that are considered to be non-routine decommissioning cases. In order to obtain a timely cleanup up these sites, NRC initiated the Site Decommissioning Management Plan (SDMP) in 1990. NRC emphasis on timely cleanup of the sites resulted from Chairman Carr's meeting with Congressman Synar on August 3, 1989, and continuing NRC concern about lack of progress at some of these sites.

Discussion:

The NRC staff [illegible] [illegible] SDMP to identify sites requiring cleanup and to provide the Commission with a status report on the actions taken to clean up the SDMP sites. In summary, the SDMP contains the following information: (1) definition of the project management plan; (2) identification of the sites requiring decontamination; (3) prioritization of the NRC's efforts in review of the contaminated sites; 4) schedule and the resources necessary to support NRC actions on contaminated site cleanup; and (5) resolution of policy and Synar hearing issues for SDMP implementation.

The SDMP not only identifies the sites requiring decontamination, it provides a description of the site, description of the wastes and activity remaining on the site, a description of the radiologic hazard from the remaining wastes and activity, the financial assurance required, the status of the decontamination activities, and the NRC proposed actions and timing to assure 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 buildings; (4) license has been previously terminated; or (5) there is contaminated or potential contamination of the groundwater from onsite wastes.

Regulations:

In 1988 the NRC promulgated the final decommission regulations under 10 CFR Parts 30, 40, 50, 70, and 72. The regulation defines decommissioning as removing a facility from service and 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 new regulation prescribes requirements for decommissioning planning, financial assurance, recordkeeping, and license termination.

Policy Issues Requiring Resolution:

There are a series of policy issues related to the cleanup 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 decontamination and decommissioning in the future. The principal policy issues that require prompt resolution for effective implementation of the SDMP are: (1) development of residual contamination criteria; (2) timeliness of cleanup; and (3) possible development of "Reopener" procedures to require additional decontamination based on changes in decontamination criteria. In addition, the SDMP also addresses several additional policy issues that need to be resolved to expedite the cleanup of the contaminated sites.

Current Status:

The NRC staff is implementing the SDMP as described in SECY-91-96 dated April, 17, 1991, and recently presented an update of the SDMP activities in SECY-91, dated October 22, 1991. The NRC staff is currently scheduled to complete an update of the SDMP in March 1992.

The SDMP program is continuing to receive increased management attention, including direct review by the Executive Director for Operations. As of the end of CY 1991, the Commission is reviewing policy issues that need resolution in order to speed progress toward completion of decommissioning. The staff is expecting to receive Commission direction in the first part of CY 1992.

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

- Over 40 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, prioritization of NRC review efforts, schedule, and resources to support NRC actions.
- Final requirements promulgated in 1988 in Parts 30, 40, 50, 70, and 72.
- Decommissioning defined as removing a facility from service and reducing the residual radioactivity to a level that will permit release of the facility for unrestricted use and termination of the license.
- Policy issues that require resolution:

  (1) development of residual contamination criteria;
  (2) timeliness of cleanup;
  (3) how to ensure finality of decommissioning;
  (4) what constitutes adequate site characterization;
  (5) how to enforce steady progress on decontamination; and
  (6) what financial incentives are there to compel cleanup.

LEVEL A SITES

CHEMETRON
CLEVELAND, OH

CHEVRON CORPORATION
PAWLING, NY

KERR-MCGEE
CIMARRON, OK

KERR-MCGEE
CUSHING, OK

SAFETY LIGHT
BLOOMSBURG, PA

TEXAS INSTRUMENTS
ATTLEBORO, MA

UNC
WOOD RIVER JUNCTION, RI

WEST LAKE LANDFILL
ST. LOUIS, MO

BP15 3/92

LEVEL B SITES

AMAX, WOOD COUNTY, WV

ARMY ARSENAL
WATERTOWN, MA

BABCOCK & WILCOX
APOLLO, PA

BP CHEMICALS
LIMA, OH

CABOT, REVERE & READING, PA

DOW, MIDLAND, MI

GSA, WATERTOWN, MA

HERITAGE MINERALS
LAKEHURST, NJ

HARTLEY & HARTLEY LANDFILL
BAY CITY, MI

MAGNESIUM ELEKTRON
FLEMINGTON, NJ

MOLYCORP, WASHINGTON, PA

MOLYCORP, YORK, PA

PESSES, PULASKI, PA

RADIATION TECHNOLOGY
ROCKAWAY, NJ

SCHOTT GLASS
DURYEA, PA

SHIELDALLOY
CAMBRIDGE, OH

WESTINGHOUSE
WALTZMILL, PA

LEVEL C SITES

ADVANCED MEDICAL SYSTEMS
CLEVELAND, OH

ALCOA, NEWBERG HIGHTS, OH

ARMY, ABERDEEN, MD

BABCOCK & WILCOX
PARKS TOWNSHIP, PA

BUDD, PHILADELPHIA, PA

CABOT, BOYERTOWN, PA

ELKEM METALS, MARIETTA, OH

ENGLEHARD CORPORATION
PLAINVILLE, MA

FANSTEEL, MUSKOGEE, OK

WYMAN-GORDON, GRAFTON, MA

MALLINCKRODT, ST. LOUIS, MO

NE OHIO SEWER DISTRICT
CLEVELAND, OH

NUCLEAR METALS
CONCORD, MA

PERMAGRAIN, MEDIA, PA

REMINGTON ARMS
INDEPENDENCE, MO

RMI TITANIUM, ASHTABULA, OH

SHIELDALLOY, NEWFIELD, NJ

VICTOREEN, CLEVELAND, OH

WHITTAKER, GREENVILLE, PA

3M COMPANY, KERRICK, MN

BP15 (3/92)

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. With such an extensive 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 NRC regulations and other requirements, 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, an assessment is made 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 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 spells out 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 those primary sanctions, the NRC has administrative mechanisms available, including Confirmatory Action Letters to state in writing a licensee's commitment to take certain immediate short-term 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 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 severity levels. A higher severity level may be assigned for cases of numerous violations that considered as a group indicate a management breakdown.

An Enforcement Conference is held for violations assessed at Severity Levels I, II, or III, and sometimes for Severity Level IV, if it is important that the NRC meet with the licensee's management. An Enforcement Conference is a meeting to discuss the violations and surrounding circumstances, causes, and the licensee's corrective actions and is normally not open to the public. 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. Others are issued directly from the Regional office.

Civil penalties are normally issued for Severity Level III or higher violations, absent mitigation, and may be issued for violations at Severity Level IV if the violations are repetitive or similar to previous Severity Level IV violations. Civil penalties may be issued for any willful violation.

If a civil penalty is to be proposed, the base value must first be determined. It 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. The 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 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.

Current Developments:

Orders may also be issued to individuals under changes to NRC's regulations that took effect September 16, 1991. 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. The 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 requirments 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. For the employer, an order might require removal or confirm removal from licensed activities, require the employer to advise prospective employers of the existence of the order when they call for reference checks about the individual, 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.

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
- Size of Civil Penalty 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 payin~ 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
- Order to individual might remove him or her from licensed activity, require notification to NRC of reemployment in licensed activities
- NRC may use Demand for Information to obtain infcrmation when considering enforcement action

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. Many controls are established to protect workers and the public from the effects of radiation.

The industry and the NRC both have roles in providing these controls and in ensuring that they are maintained. The NRC establishes rules, 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 plant lies with the licensee. The NRC inspection program is designed to conduct selective examinations to ensure that the licensee is meeting his prescribed responsibility. The NRC inspection program is audit oriented; thus, it does not examine every activity or item, but verifies, through carefully selected samples, that the activities are being properly conducted or operated to enhance or ensure safety. What to sample, the sizes of the samples, and the frequencies of the inspection efforts are based on the importance of the activity or system to overall safety and available resources. The inspection program is preventive in nature and is designed to anticipate and preclude significant events and problems by identifying underlying issues. The inspection process, from a systems approach, monitors the licensee's activities and provides feedback to the licensee's plant management to allow it to take appropriate corrective actions. However, implementation of the NRC inspection program does not supplant the licensee's programs or responsibilities. Rather, the inspection program provides a feedback mechanism and an independent verification of the effectiveness of the licensee's implementation of its programs to ensure that operations are being conducted safely and in accordance with applicable NRC requirements.

Inspections are performed on power reactors under construction, in test, and in operation. The inspections are conducted primarily by region-based and on site resident inspectors. Resident inspectors are stationed at each reactor under construction and in operation. Region-based inspectors operate from the five regional offices located in or near Philadelphia,

Atlant, Chicago, Dallas, and San Francisco. These programs aresupplemented by inspections conducted by special teams comprised of personnel from both headquarters and regional offices.

Inspections are part of NRC's review of applications for licenses as well as NRC's 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 activities related to design, procurement and the plans for fabrication and construction.

During construction, a sampling of licensee activities is inspected to make sure that the requirements of the construction permit are followed and that the plant is built according to design and applicable codes and standards. Construction inspections look for qualified personnel, quality material, conformance to approved design and for a well-formulated and satisfactorily implemented quality assurance program for ensuring the quality of construction.

As construction nears completion, preoperational testing to demonstrate the operational readiness of the plant and its staff begins. Inspections during this phase determine whether the licensee has developed adequate test plans, assure that tests are consistent with NRC requirements, and determine that the plant and its staff are prepared for safe operation. Inspections during the preoperational phase involve (1) reviewing overall test management procedures; (2) examining selected test procedures for technical adequacy; and (3) witnessing and reviewing selected tests to determine their outcomes and the consistency of planned and actual tests. In addition, inspectors review the qualifications of operating personnel and assure that operating procedures and quality assurance plans are developed and implemented.

About six months before the operating license is issued, a startup phase begins in preparation for 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 preoperational testing, NRC inspection emphasis is placed on test management procedures and results. The licensee's management system for startup testing is examined, test procedures are analyzed, tests are witnessed, and licensee evaluations of test results are reviewed.

When startup testing is completed satisfactorily, routine operations begin. Thereafter, NRC continues its inspection program throughout the operating life of the plant.
The responsibility for safe operation of nuclear plant lies with the licensee. The NRC inspection program is designed to conduct selective examinations. An onsite 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 office's determination of what additional inspections 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 reactor core physics. The specialist inspectors provide a perspective that is different from, but complementary to, that of the resident who, of necessity, is a generalist. Since the specialists inspect many different plants, they see many different ways of accomplishing a function, and they develop a more comprehensive view of their specialties.

The inspection program for operating reactors is defined in the NRC Inspection Manual in terms of frequency, scope, and depth. Detailed inspection procedures provide instructions and guidance for inspectors. The inspection program is comprised of three major elements: core inspections, the minimum done at all plants; area of emphasis inspections, special inspections to focus on a specific issue; and discretionary inspections, those which are required to adequately resolve safety issues identified by other inspections or as a result of plant events. The program is structured to ensure that the finite resources available for inspection are used efficiently and effectively with increased attention being devoted to those plants where, based on licensee performance, improvements in safety may be needed.

The inspection program is an important element in the NRC's regulatory program. Its results are factored into NRC's overall evaluation of licensee performance under the Systematic Assessment of Licensee Performance (SALP) program designed to ensure that nuclear power reactors are constructed and operated safely and in compliance with regulatory requirements. When a safety problem or failure to comply with requirements is identified, NRC directs prompt corrective action by the licensee, and takes, as necessary, appropriate enforcement action.

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

Contact:

Mark Peranich, Chief, Inspection Program Development Section,
Inspection and Licensing Program Branch
US NRC, Washington, DC 20555 (301) 504-3078

HIGHLIGHTS OF INSPECTION PROGRAM

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

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

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

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

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

- o Inspection program defined in NRC Inspection Manual and detailed inspection 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) discretionary inspections, those which are required to adequately resolve safety issues.

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

ENVIRONMENTAL EFFECTS 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, biota, and terrestrial and aquatic organisms. Most operating nuclear power plants have a Final Environmental Statement (FES) issued by the NRC which details the potential impacts resulting from the routine operation of the 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 not only keep releases of radioactive material to unrestricted areas during normal operation as low as is reasonably achievable, but also complies with the applicable provisions of 10 CFR 20.106, "Radioactivity in Effluents in Unrestricted Areas". 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 Effects:

Radiation releases and their associated doses are reported by most licensees in Semi-Annual Effluent Release Reports while radioactivity levels in various environmental media are reported in the Radiation Environmental Monitoring Program (REMP) Reports. The former report includes 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 usually compared to the design objective doses of Appendix I. 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 REMP report 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 a 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 such as broad leaf vegetation. The results of the REMP 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 1960's) were required to meet.

On December 3, 1970, the Atomic Energy Commission, the predecessor Agency to the NRC, established new regulations, 10 CFR 50.34a and 10 CFR 50.36a, which specified the design and operating requirements for nuclear power reactors to keep levels of radioactivity in effluents "as low as practical." These regulations provided qualitative but not numerical criteria for determining when design objectives and operations meet the specified requirements. 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 criteria "as low as practical." 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 Semi-Annual Effluent Release Reports and the REMP Reports, 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 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 aerial surveillance of nuclear power plants in which measurements of direct radiation are made.

The results of these monitoring efforts are documented in the licensee's Semi-Annual Effluent Release Reports and the REMP Reports. The NRC documents the results of its independent monitoring and assessment efforts in reports such as NUREG/CR-2907, "Radioactive Material Released from Nuclear Power Plants," NUREG/CR-2850, "Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites," and NUREG-0837, "NRC TLD Direct Radiation Monitoring Network."

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. A significant contributor to the reduction in airborne effluents has been the addition of the Augmented Offgas Systems to boiling water reactors.

Contact:
Lemoine Cunningham, Chief, Radiation Protection Branch
US NRC, Washington, DC 20555 (301) 504-1086

ENVIRONMENTAL EFFECTS 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 semi-annual effluent release report.

- Licensees report radioactivity levels from the environmental sampling of air, ground, water and ingestion pathways in the radiation environmental monitoring program (REMP).

- 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, and also comply with 10 CFR 20.106.

- 10 CFR 20.106 outlines the radioactivity in effluents in unrestricted areas.

- 10 CFR 50.34a & Appendix I provide numerical guides for design objectives and limiting conditions for operation to meet as low as practical criteria.

- 40 CFR 190 outlines standards for effluents for uranium fuel cycle licensees.

- The staff verifies that licensees evaluate potential impacts through a mobile lab, state contracts, interlaboratory comparison, and aerial surveillance.

- The staff documents its independent monitoring and assessment effort in NUREG-2907, NUREG-2850, and NUREG-0837.

- Effluents released have decreased significantly.

Uranium Mill Tailings

Background:

A lack of orders for new nuclear power plants and the importation of uranium from other countries has 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 inactive mills (formally used in providing uranium for the weapons program) and from active mills regulated by the NRC 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 constituent in uranium mill tailings is radium, which is radioactive and 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, safety, 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 to NRC. Under Title I, the NRC must evaluate and concur that the Department of Energy's (DOE) 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 hazard; (2) EPA authority to set generally applicable standards for both radiological and nonradiological hazards; and (3) eventual State or Federal ownership, under license to 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:

The 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, it is believed that by use of the standards and NRC's implementing regulations tens of thousands of radon-related lung cancer deaths will be avoided.

Current Status

Title I -- Reclamation Work at Inactive Tailings Sites

Twenty-four inactive mill tailings 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 in 1991 and is estimated to be completed by 2020 at a cost of over $500 million. DOE has completed remedial action on approximately half of the Title I sites. It is anticipated that in 1992 DOE will become an NRC licensee for the long-term custodial care of the completed mill tailings sites located near Shiprock, New Mexico and Spook, Wyoming.

Title II -- Licensed Mill Tailings Sites

Of 27 NRC licensed uranium recovery facilities 14 are either expected to begin, or have already started, reclamation activities to provide long-term stabilization and closure of the tailings impoundments. These NRC-licensed sites are located in New Mexico, Utah, Wyoming. There also are 6 uranium mills in Agreement States (Colorado, Texas, and Washington) that have similar non-operational tailing impoundments.

In the fall of 1991, NRC, EPA, and the affected mill tailings Agreement States had intensive discussions concerning the non-operational tailings impoundments at these facilities and EPA's duplicative requirements under the Clean Air Act (CAA). All parties agreed that there was a need to eliminate the dual regulation created by NRC's authority under UMTRCA and Atomic Energy Act and EPA's authority under CAA. This interagency consultation resulted in the execution of a Memorandum of Understanding (MOU). The primary purpose of the MOU is to ensure that non-operational uranium mill tailings piles licensed by NRC or an affected Agreement State achieve compliance with EPA's standards as expeditiously as practicable. A guiding objective in the MOU is that such closure occur by the end of 1997 for the current licensed non-operational disposal sites. The MOU specifies that the final closure shall be enforceable by NRC or the affected Agreement States.

Contact:

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 abandoned sites, with ultimate Federal ownership under license to NRC.

- Title II of UMTRCA provided: (1) NRC authority to control radiological and nonradiological hazard; (2) EPA authority to set generally applicable standards for both radiological and nonradiological hazards; and (3) eventual State or Federal ownership, under license to 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 program is estimated to cost approximately $1.3 billion and to be completed by 1998. The DOE groundwater cleanup phase was initiated in 1991 and is estimated to be completed by 2020 at a cost over $500 million.

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

- There are 14 NRC licensed and 6 Agreement State-licensed facilities that have non-operating mill tailings impoundments.

- An NRC-EPA-Agreement State MOU of October 1991, provides that the goal for closure of the tailings impoundments at these sites shall occur by the end of 1997.

STATE COMPLIANCE WITH 1993 AND 1996 MILESTONES 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 towards being able to manage their low-level radioactive waste (LLW) by 1993. However, slow progress by the States in developing new LLW disposal facilities will result in the storage of LLW at many generator sites beginning January 1, 1993. This paper includes background information on the LLRWPAA and the status of new low-level waste (LLW) disposal facility development. Related regulatory actions are summarized including a proposed rulemaking which would establish criteria for on-site storage of LLW after January 1, 1996.

Low-Level Waste Disposal:

Low-level radioactive waste (LLW) is a general term for a variety of radioactively contaminated waste 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 1.4 million cubic feet of LLW was disposed of in 1991. LLW is currently disposed of using shallow land burial at privately operated facilities located in the States of Nevada, South Carolina, and Washington. However, on January 1, 1993, these sites are expected either to close or stop accepting waste from outside their regions.

The LLRWPAA requires the sited States of Nevada, South Carolina, and Washington to make disposal capacity available to LLW generators until December 31, 1992, subject to: the States and compacts meeting the other milestones of the LLRWPAA, the sites remaining operational, and received waste being within site specific volume limitations. While the Washington facility is scheduled to remain open serving its compact (Northwest) on January 1, 1993, the Nevada and South Carolina facilities are scheduled to close on December 31, 1992. At present, nine compacts have been formed representing 42 States. Figure 1 shows the current arrangements of compacts and unaffiliated States (i.e., those States not in a compact).

To help ensure 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. Only the States of California, Illinois, and Nebraska met the January 1, 1992, milestone requirement to submit a facility license application. The remaining milestones are:

- January 1, 1993 - If a State or compact cannot provide for disposal of its LLW after January 1, 1993, generators can request the State to take title to and possession of the generated waste. The State also becomes liable for damages as a consequence of failure to take possession of the waste. In 1993, States may avoid taking title and possession of the waste and assuming liability, but will forfeit the surcharge rebates established by the LLRWPAA.

- January 1, 1996 - The States, upon proper notice by the generator or owner, shall take title to and be obligated to take possession of LLW. The State will also be liable for all damages directly or indirectly incurred by the generator or owner if it fails to take possession as soon after January 1, 1996, as the generator or owner notifies the State that the waste is available for shipment.

Table 1 shows the dates by which compact host States and unaffiliated States accomplished, or expect to accomplish, key steps in developing new disposal facilities.

Since new LLW disposal facilities are not expected to be operational by January 1, 1993, and the existing LLW disposal sites are expected to either close or stop receiving LLW from outside their regional compacts on January 1, 1993, many licensees who generate LLW must store their LLW on site until disposal capacity is available unless other arrangements for storage or disposal can be made. Nearly all the Governors' Certifications submitted to meet the 1990 milestone of the LLRWPAA indicated the State planned on interim storage by waste generators during the 1993 through 1996 period.

Regulatory Actions:

Slow progress in some States and compacts towards meeting the January 1, 1996, milestone of the LLRWPAA has raised Commission concerns associated with on-site storage of low-level radioactive waste. While the public health and safety can be adequately protected if low-level radioactive wastes are stored, the public health and safety will be enhanced by disposal, rather than long-term, indefinite storage of wastes. Disposal of wastes in a limited number of facilities licensed under existing regulations (10 CFR Part 61) will provide better protection of the public

health and safety and environment than storage at hundreds of sites around the country. Permanent disposal of low-level radioactive waste has always been the preferred option for managing wastes as reflected in the LLRWPAA. Because of these concerns and as a result of the Commission's consideration of the staff's analysis in SECY-91-306, NRC is proposing to amend its regulations to establish requirements for on-site storage of LLW by licensees after January 1, 1996.

In this proposed rulemaking, the Commission is restating and emphasizing its position that it will not look favorably upon on-site storage of low-level radioactive waste by generators after January 1, 1996, the final milestone of the LLRWPAA. The Commission considers on-site storage to be a least desirable measure. Under the proposed amendments, on-site storage of low-level radioactive waste 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 could document that it has exhausted other reasonable waste management options. Such options include:

- Requesting the State in which the generator of the waste is located to take title to and possession of the waste in accordance with the LLRWPAA, and

- 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, 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:

The Commission has requested that the staff prepare and submit the proposed rulemaking on storage of LLW beyond 1996 to the Commission for consideration and approval by May 1, 1992. It is the Commission's strong desire to have the final rule in place by December 31, 1992.

The U.S. Supreme Court is reviewing the decision of the U.S. Court of Appeals for the Second Circuit in *New York v. United States*. The State of New York asserts that the LLRWPAA, especially the take title provision, exceeds the limits imposed on the Federal Government under the Tenth Amendment of the Constitution. Oral arguments were heard by the Supreme Court on March 30, 1992, and a decision is expected by July 1, 1992. Should the United States not prevail, there are three possible

adverse outcomes. The Supreme Court may find only the take-title provision of the LLRWPAA unconstitutional, allowing the remainder of the LLRWPAA to stand. The Supreme Court may find the entire LLRWPAA to be unconstitutional. The Supreme Court may also find that the take-title provision is constitutional as it applies to compacts, and unconstitutional as it applies to unaffiliated States. Any of these findings could adversely affect the development of new disposal facilities resulting in an increased reliance on on-site storage of LLW.

Contact:

Paul H. Lohaus, 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

Table 1
Actual and Estimated Dates for
Completing Steps in Facility Development

Compact/Host State	Select Site	Submit License Application	Operate Facility
Appalachian/Pennsylvania	Spring 1994	Spring 1994	Fall 1996
Central/Nebraska	Dec 1989	Jul 1990	Fall 1995
Central Midwest/Illinois	Early 1992	May 1991	Late 1993
Midwest/Ohio	Unscheduled	Unscheduled	Unscheduled
Northeast/Connecticut &	Late 1992	Jun 1994	Late 1996
New Jersey	Early 1993	Jan 1995	Early 1997
Southeast/North Carolina	Oct 1993	Nov 1993	Feb 1996
Southwest/California	Mar 1988	Dec 1989	Jan 1993
Unaffiliated States			
Maine	1994	Late 1995	Late 1997
Massachusetts	Jun 1994	Nov 1994	Dec 1996
New York	Unscheduled	Unscheduled	1998
Texas	Feb 1992	Mar 1992	Mid 1996
Vermont	Jan 1995	Oct 1997	Aug 1999

LOW-LEVEL RADIOACTIVE WASTE COMPACT STATUS

MARCH 1992

*Operating LLW Disposal Sites

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

HIGHLIGHTS

State Compliance with 1993 and 1996 Milestones of the
The Low-Level Radioactive Waste Policy Amendments Act of 1985

- LLRWPAA established milestones, incentives, and penalties for States to develop new LLW disposal facilities
  - Milestones established in 1986, 1988, 1990, 1992, 1993, and 1996
  - Waste disposal surcharges and take-title and possession provisions are penalties for failure to comply
  - Partial rebate of surcharges to States provide incentive in the form of financial assistance
- Majority of States met the first three milestones.
- Only three States (California, Illinois and Nebraska) met the 1992 milestone and only these States are expected to meet the 1993 and 1996 milestones.
- Existing disposal facilities are expected to close or stop receiving LLW from outside their compacts on January 1, 1993.
- On-site storage of LLW at many generator sites is expected after January 1, 1993 due to lack of access to disposal facilities.
- Existing NRC guidance in conjunction with current regulations provide regulatory and licensing framework for LLW storage.
- While 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 on on-site storage of LLW. NRC considers on-site storage to be the least desirable measure.
- Commission has directed the development of a proposed rulemaking that would establish criteria for on-site storage of LLW after January 1, 1996. Licensees would have to exhaust all other waste management options before storing LLW on-site. Options include:
  - Request State to take title to and possession of the LLW.

  - Contract, 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.

- Proposed rulemaking due to the Commission for consideration and approval by May 1, 1992. Goal - final rule in place by December 31, 1992.

- U.S. Supreme Court considering constitutionality of LLRWPAA; decision expected by July 1, 1992.