OFFICE OF NUCLEAR REACTOR REGULATION
U.S. NUCLEAR REGULATORY COMMISSION

NRR

TECHNICAL NEWSLETTER

NUREG/BR-0125
Vol. 4, No. 1
May 1992

A Tribute to

Frank J. Witt

1931 – 1992

Frank J. Witt joined the NRC as a chemical engineer in 1976 after having spent 23 years working for the General Electric Company. Actively involved in the nuclear power field at GE, Frank spent considerable time with the Knowles Atomic Power Division. Many of GE's nuclear chemical engineering systems in use today evolved from Frank's pioneering work.

Frank participated in all of the chemical engineering functions at NRC during his too-brief tenure. He headed up the effort on post accident sampling systems (PASS) and represented the agency at the Tenth Annual PASS Owners Group Meeting as a keynote speaker. Frank was also responsible for the full system chemical decontamination programs at the NRC, coordinating the NRC teams in dealing with the Westinghouse Owners Group, the General Electric BWR Owners Group, and the Combustion Engineering Owners Group. Frank was responsible for introducing hydrogen water chemistry into PWRs and BWRs for control of intergranular stress corrosion cracking (IGSCC). Frank was also considered to be the NRC expert on primary and secondary water chemistry in PWRs. He was the NRC expert on contaminants in diesel generator fuels and lubricants and the effect of these contaminants on diesel generator materials. Frank knew about contaminants in plant piping thermal insulation and was the NRR representative for revising Regulatory Guide 1.36 and ASTM Standard C-16 on thermal insulation. He had responsibility for examining microbially influenced corrosion (MIC) in nuclear power plant water systems and in monitoring methods to avoid MIC. Frank was active in a number of technical associations including the National Association of Corrosion Engineers, the American Society of Testing and Materials, and the Electric Power Research Institute. He was also on the ASTM Core Committee for Nuclear Grade Coatings.

Frank's friends and colleagues at NRC and throughout the nuclear power industry will miss him personally and his ever-present contributions to the chemical engineering field will be missed by engineers who knew him only through his work.

Frank leaves his wife, Louise, and three children, who were fortunate to have had a husband and father who made an important contribution to this country.

Thomas E. Murley

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An Introduction To Non-Power Reactors

Alexander Adams, Jr.

Non-Power Reactors, Decommissioning and Environmental Projects Directorate

Because of their low power levels and inherent safety features, the 61 NRC-regulated non-power reactors (NPRs) are located in urban areas. In fact, the majority of NPRs are located on university campuses and are used for training and research. NPRs in this area are located in Gaithersberg at the National Institute of Standards and Technology, in Bethesda at the Armed Forces Radiobiology Research Institute on the grounds of the Naval Hospital, in College Park at the University of Maryland, and in the District of Columbia at Catholic University. The power levels of NRC-licensed NPRs range from 0.1 watt to 20 MW.

NPR types are primarily distinguished by their fuel design. The major classes of NPRs are described below.

(1) Aerojet-General Nucleonics (AGN) Reactor (Figure 1)

The AGN has the simplest design. It is a compact, low-power (0.1 to 5 watts), self-contained homogeneous core design with low-enriched (20% U-235) powdered uranium oxide fuel embedded in a polyethylene moderator. A graphite neutron reflector and a lead and water shield surround the 10-inch diameter core. The control rods consist of fuel instead of a neutron poison. They enter the core from the bottom and are inserted to bring the reactor to critical and withdrawn to shut down the reactor. A core thermal fuse is used to hold the lower and upper sections of the core together. Because the core thermal fuse contains fuel with greater uranium density than the rest of the core, it is the hottest part of the core. The core thermal fuse is designed so that its polyethylene moderator will melt if an accident occurs, causing the upper and lower sections of the core to separate and providing a backup shutdown mechanism.

(2) Critical Experiment Facility

One critical experiment facility at the Rensselaer Polytechnic Institute (RPI) remains licensed by NRC. This was once a very common reactor type, used to acquire the many basic measurements of critical fuel behavior necessary to support nuclear research and power reactor design. It is a very low-power (100 watts) reactor capable of many core configurations. The reactor core sits in a pool of light water that acts as coolant, moderator, and reflector. In addition to control rods, the pool can be quickly drained to remove the moderator from the reactor and shut it down.

(3) Argonne Nuclear Assembly for University Training (Argonaut) Reactor (Figure 2)

These are low-power (10 kW to 100 kW) reactors with high-enriched (93% U-235) fuel contained in aluminum clad Materials Testing Reactor (MTR) type plates. The fuel is placed in fuel boxes through which the cooling water flows. The water in the fuel boxes and graphite surrounding the fuel boxes act as the moderator. Semaphore-type control rods swing in between the fuel boxes. Massive concrete blocks weighing several tons shield the reactor and must be unstacked to gain access to the core.

(4) Training Reactor, Isotopes Production, General Atomics (TRIGA) Reactor (Figure 3)

This is the most common NPR design. These are low- to medium-power reactors (20 kW to 1500 kW) with U-ZrH fuel-moderator (either 20% or 70% U-235 enriched) in the form of pins clad with aluminum or stainless steel. The reactor is moderated by water and the hydrogen in the fuel. TRIGA reactors are either water or graphite reflected. The core sits at the bottom of an open pool and is cooled by natural convection. The reactor can be safely pulsed to very high power levels for very short periods by pneumatically ejecting a specially designed control rod from the reactor core. TRIGAs have been routinely pulsed with reactivity insertions of 5.00$ resulting in pulse power greater than 4000 MW. The pulse is shut down by a strong prompt negative fuel temperature coefficient. As the fuel heats up, the hydrogen atoms in the fuel-moderator vibrate and transfer energy to the neutrons in the fuel, increasing the probability that they will leave the fuel without causing fission to occur. These reactors also use Doppler broadening of neutron absorption as the fuel heats up to shut down a pulse.

This is the only U.S. NPR design supported by the manufacturer and available for sale.

(5) PULSTAR

The PULSTAR reactor from American Machine and Foundary (AMF) Atomics also was designed as a pulsing reactor. These are medium- to high-power (1 MW to 2 MW) reactors with low-enriched (4% to 6% U-235) uranium-dioxide pin fuel clad in zirconium. The fuel elements resemble small pressurized water reactor fuel elements. The two PULSTAR reactors no longer pulse because of the cost of maintaining pulse equipment and operator proficiency. Like the TRIGA, pulsing was initiated by pneumatically ejecting a pulse control rod from the reactor core. The pulse was shut down by the Doppler broadening of neutron absorption as the fuel heated up. The core sits at the bottom of an open pool and is cooled by forced convection.

(6) Tank Reactor (Figure 4)

This is normally a high-power (5 MW to 20 MW) reactor with plate-type high-enriched (93% U-235) aluminum clad MTR-type fuel and forced convection cooling. Heavy water may be used as a coolant and moderator or as a reflector. To gain excess to the reactor core, which is in a sealed tank, the tank top must be removed. Although all tank reactors do not use heavy water as a coolant, the tank was an original design feature to keep the heavy water system sealed from the environment. This prevents the loss of very expensive heavy water and also prevents the heavy water from absorbing light water from the air. In some cases, a decrease in heavy water purity to 99.3% will result in a reactor that cannot be made critical. Heavy water is used because it is an ideal moderator and has a very low neutron absorption cross section. This allows the core to be designed with large gaps between fuel elements that can accommodate experiments. Heavy water reactors also have very high thermal-to-fast-neutron ratios.

(7) Plate-Type Pool Reactors (Figure 5)

These reactors have a wide variety of power levels (0.1 watt to 10 MW) with plate-type aluminum-clad fuel. Originally operated with high-enriched fuel, a number of plate reactors have been converted to low-enriched fuel as part of the Government's program to reduce the amount of high-enriched uranium at NPRs. The reactor core sits at the bottom of a open pool and is cooled by forced convection. These reactors are used for a variety of purposes, for example, beam experimentation, isotope production, neutron radiography, and neutron activation analysis. A variant on this design, the core of the reactor at the University of Missouri in Columbia is in a pressure vessel that sits in an open pool. The control rods are outside the reactor core and pressure vessel and adjust the amount of neutrons that reach the external reflector and reflect back into the core.

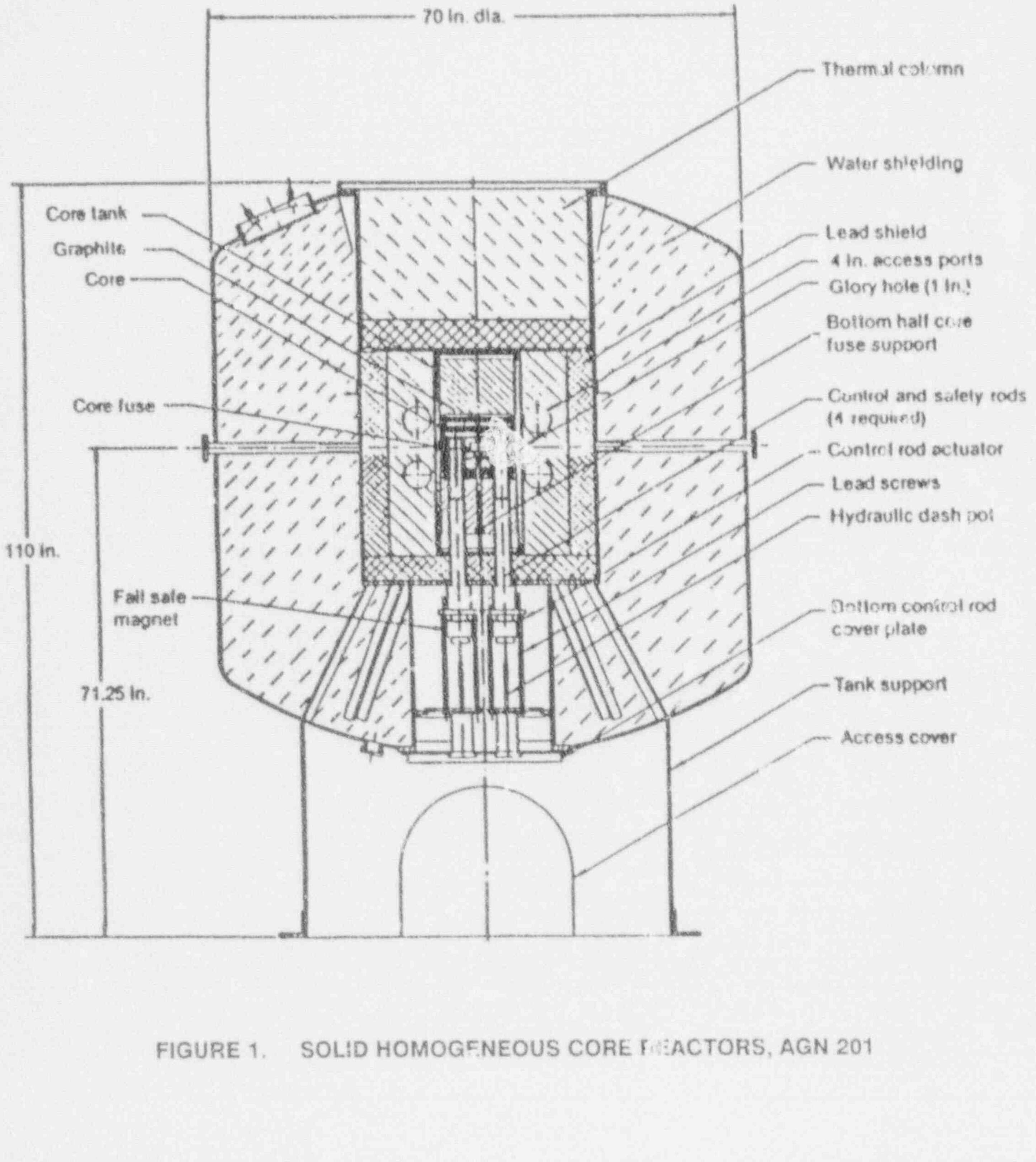


FIGURE 1. SOLID HOMOGENEOUS CORE REACTORS, AGN 201

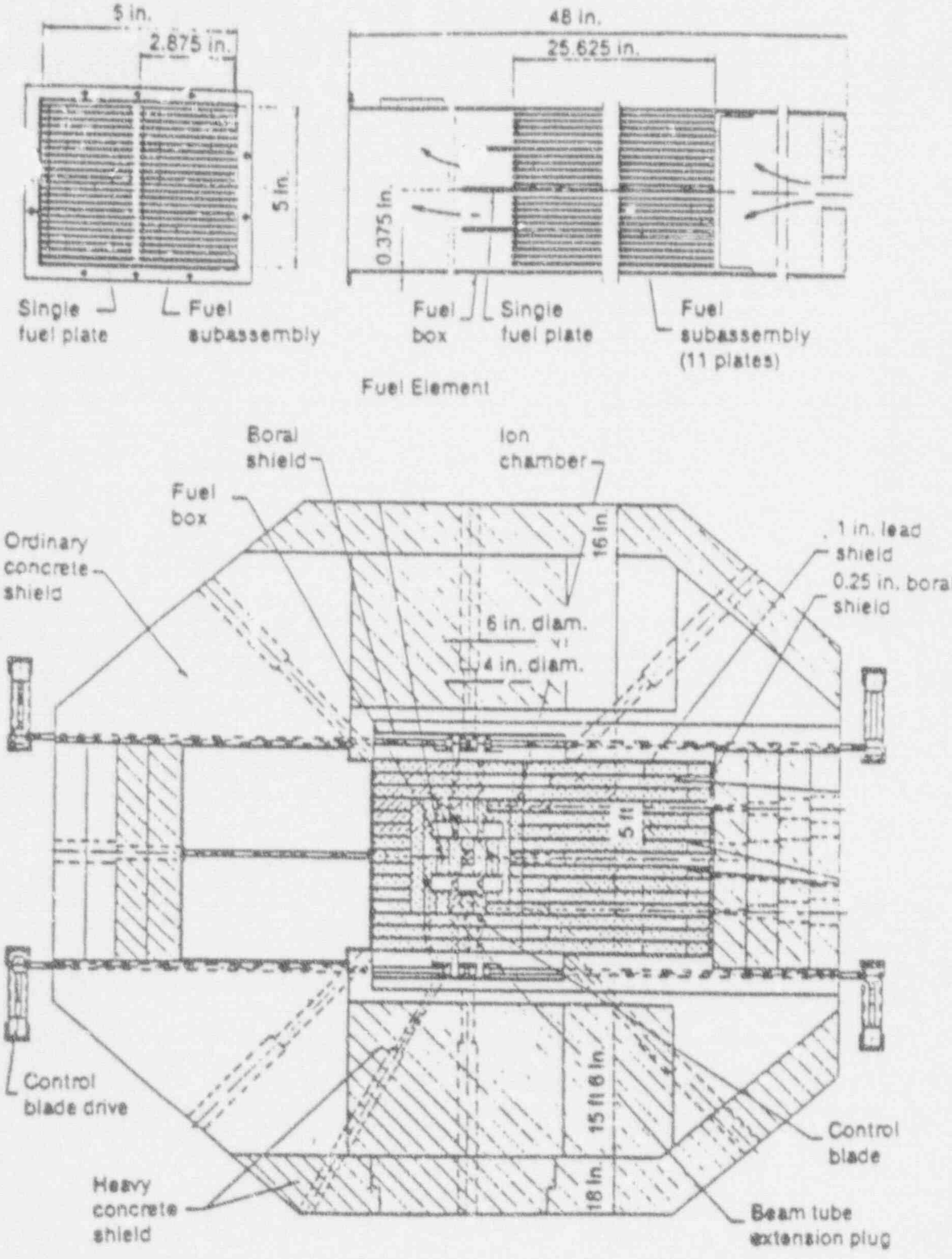


FIGURE 2. HORIZONTAL SECTION OF A NON-POWER ARGONAUT REACTOR

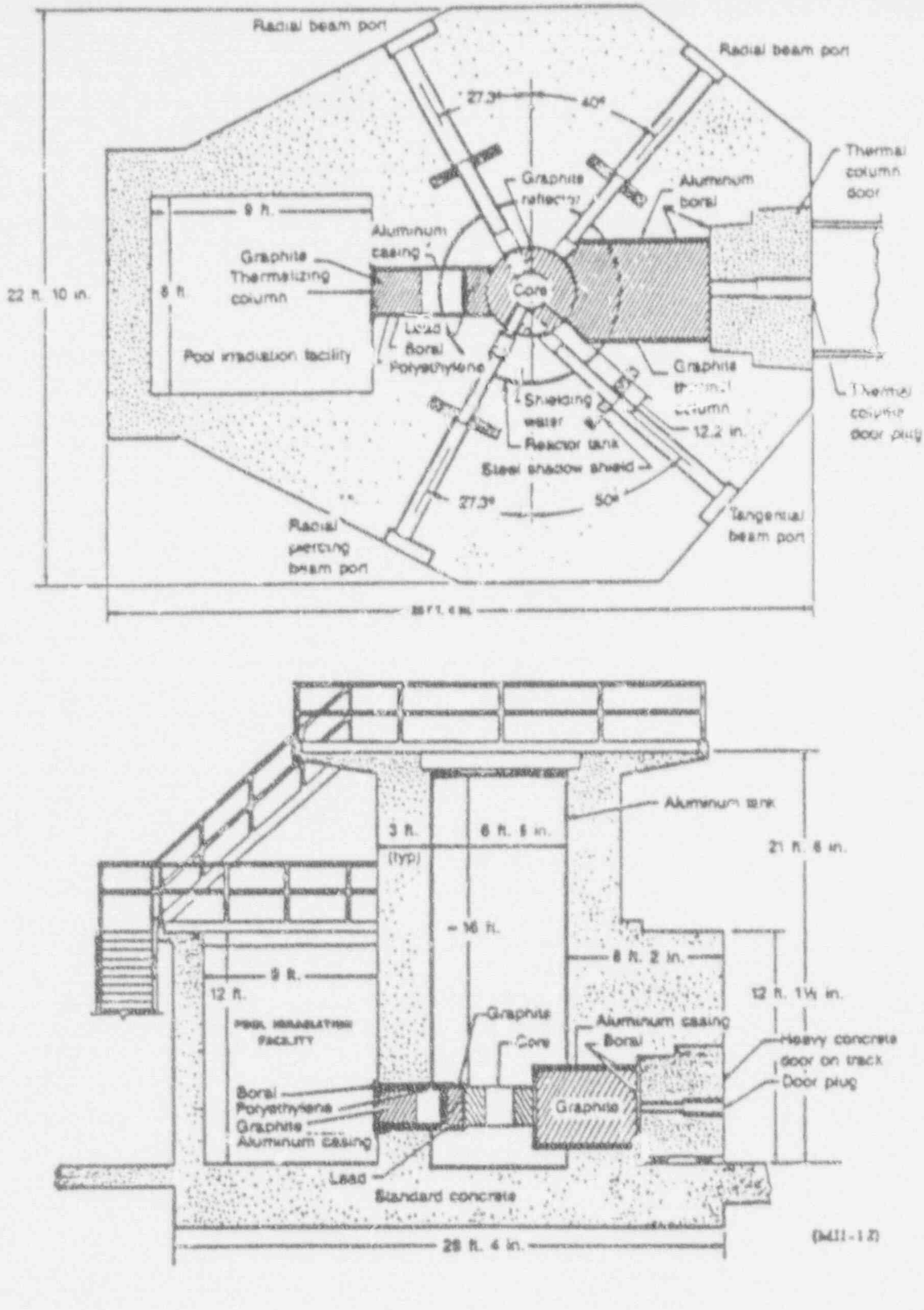


FIGURE 3. 250 KW TRIGA MARK II WITH POOL IRRADIATION FACILITY, THERMAL COLUMN, AND TANGENTIAL BEAM PORTS

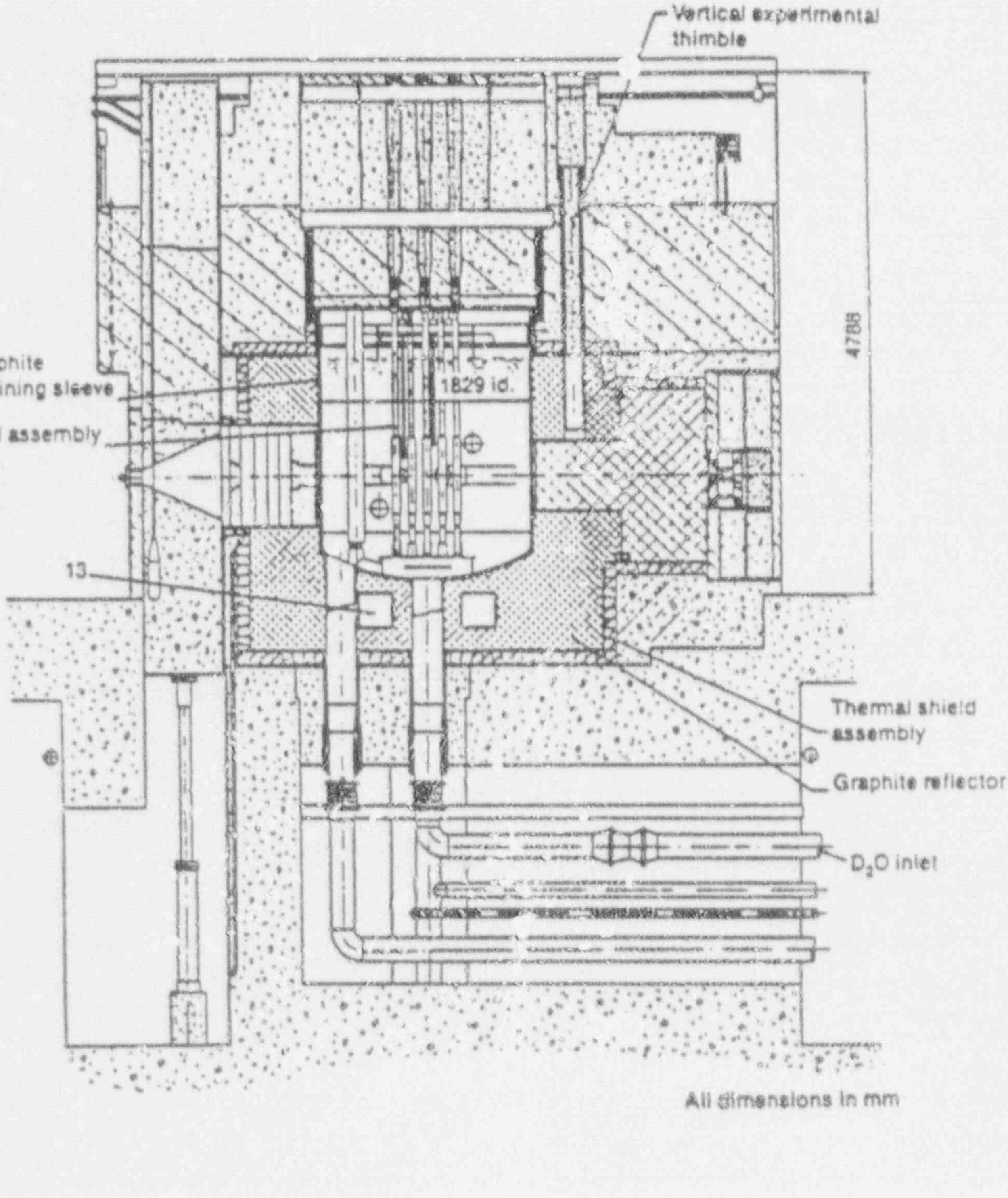


FIGURE 4. VERTICAL SECTION OF A NON-POWER TANK REACTOR

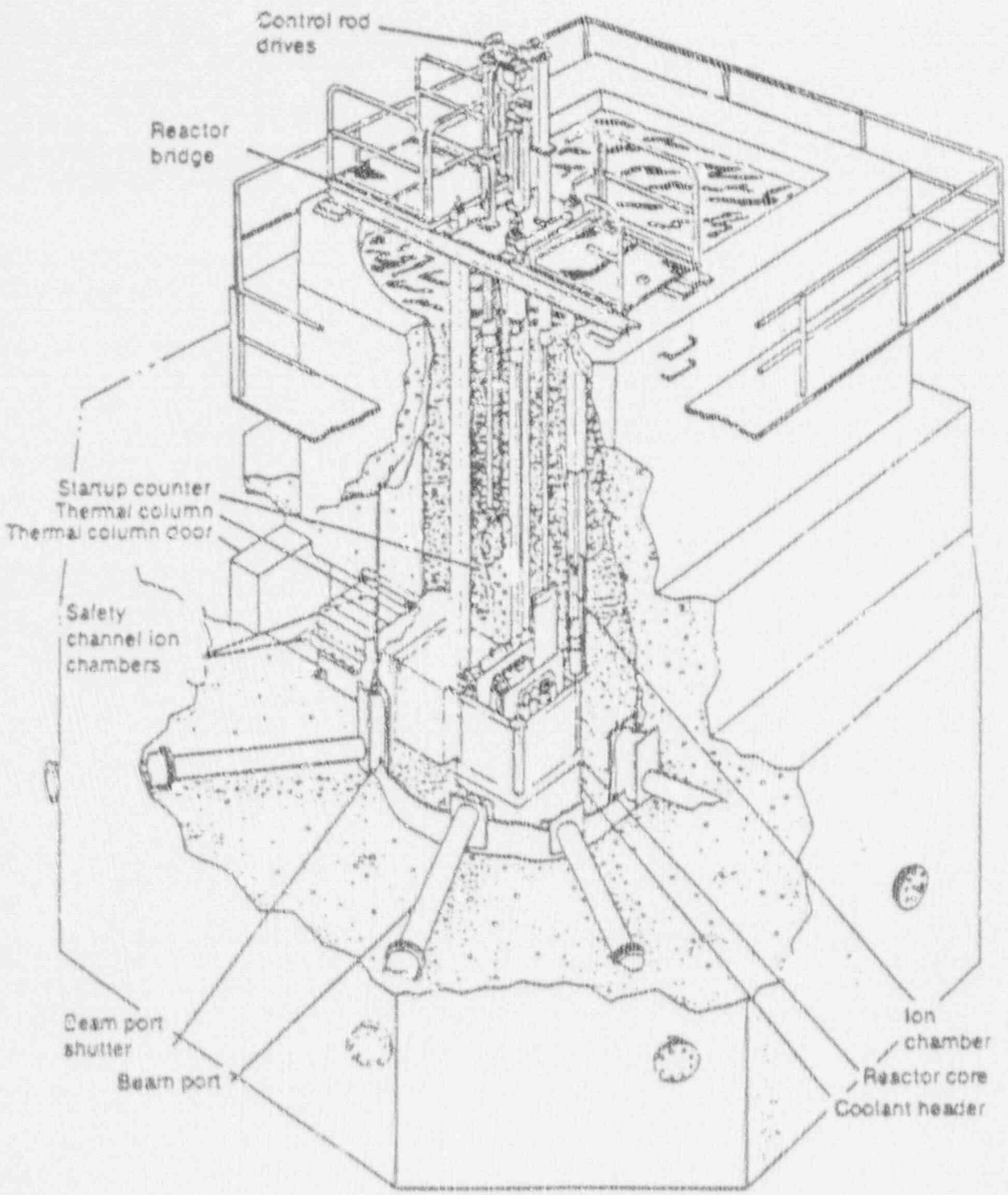


FIGURE 5. OPEN POOL REACTOR

Inspections, Tests, Analyses, and Acceptance Criteria

Tom Boyce, PDST

Introduction

The Standardization branch of the Division of Advanced Reactors and Special Projects has been coordinating the Office of Nuclear Reactor Regulation's (NRR's) review of several new reactor designs which are intended to be licensed under the new licensing process for a facility established under 10 CFR Part 52. These new designs, together with Part 52, will provide the foundation for the next generation of reactors to be licensed in the United States. The inspections, tests, analyses, and acceptance criteria (ITAAC) have emerged as a distinctive part of the design reviews because of the unique role which they play in this new licensing process. The concept of ITAAC is also new, and thus has been a topic of considerable interest to industry, the Commission, and the staff during the course of the design reviews.

Under the 10 CFR Part 52 "one-step" licensing process, the NRC must be able to make all safety findings for a facility before the first shovel breaks ground for construction. This is no small challenge. In contrast, the plants currently operating were licensed under the "two-step" licensing process under 10 CFR Part 50, whereby the NRC would first grant a utility a construction permit to build a facility and then issue it a second license to operate the facility after it was built. Under the Part 52 process, the NRC would issue a single combined license (COL) for a facility to be both built and operated. However, to confirm that the facility was built satisfactorily prior to its actual operation, the concept of ITAAC was developed. ITAAC consist of various inspections, tests, and analyses which the utility performs for significant aspects of the facility, and then measures the results against prescribed acceptance criteria. For example, if the flow rate of a pump has particular safety significance for the design, the applicant would perform various inspections, tests, or analyses and compare the results to acceptance criteria in order to verify the flow rate. Prior to fuel loading, the ITAAC provide the basis for the Commission to confirm that the plant was built and will operate in accordance with the approved design in the COL.

Part 52 Design Review And Certification Process

Under Part 52, after the staff performed an extensive review of a design and issued a final safety evaluation report, the Commission would issue a final design approval (FDA) for the design. The Commission would then make a rule certifying the design and the ITAAC for the design in order to provide final resolution of all design issues, making them no longer subject to litigation. The certified design can then be referenced by an applicant for a COL. Since the design is not subject to litigation, this will streamline the process for licensing new facilities. In addition, when referenced by multiple licensees, a certified design would become the standardized design for future nuclear plants.

Status Of ITAAC Review

The General Electric Company (GE) has been selected to be the lead plant for which to develop the first ITAAC during the design review for the Advanced Boiling Water Reactor (ABWR). In 1991, the staff held numerous meetings with GE and NUMARC on this subject, which led to the development of a set of "pilot" ITAAC which were initially submitted for the staff to review in September 1991. A full ITAAC submittal is expected in early 1992 after agreement is reached on the pilot ITAAC. Staff approval of the full ITAAC submittal is scheduled for the fall of 1992.

Types Of ITAAC

GE is developing several types of ITAAC for the GE ABWR. ITAAC are being developed for the approximately 140 systems of the ABWR design. Examples of these "system" ITAAC are the ITAAC requirements for systems such as the residual heat removal system and the reactor protection system. GE also is developing "generic" ITAAC for issues that apply to many systems, such as those for the environmental qualification of key components. GE has identified several areas as design interfaces with the rest of the facility. These include site-specific elements such as the service water intake structure and the ultimate heat sink. A utility referencing the design will be required to meet these interface requirements by submitting "facility" ITAAC that will be provided in the application for a COL.

GE provided design information having a level of detail that was insufficient for making a final safety determination in certain areas of the review. GE did not provide a high level of detail in these areas because they are areas of rapidly changing technology or because they represent as-built or as-procured type information. These areas include instrumentation and controls, piping design, radiation protection, and control room design. GE is developing design acceptance criteria (DAC) specifying the requirements which must be met for future design work in these areas.

Review Issues

1. **Level of Detail And Extent of Standardization.** The level of detail in the certified design and the ITAAC will greatly affect the extent of standardization in facilities referencing the certified design. During the review process, a graded approach to the certified design information based on its safety significance has been taken to provide the appropriate safety benefits of standardization.

2. **Separation of Tier 1 and Tier 2 Information.** The rule certifying a design will contain a sufficient level of design detail so that the rule provides for a standardized design and provides for early resolution of design issues, while allowing the flexibility to accommodate necessary changes to a facility during its construction and operating lifetime. Thus, the rule will only certify a selected portion of the information (called Tier 1 information) submitted by the designer. This portion of the design information would be verified by the ITAAC and would have very high thresholds for changes. The remainder of the design information, typically that contained in the standard safety analysis report (SSAR), would be controlled by a "50.59-like"

process, and would include a requirement to ensure that the change did not affect any Tier 1 information.

3. **Overlap With Traditional Inspection Process Under Part 50.** Some areas of ITAAC may overlap with the existing regulations of 10 CFR Part 50 and the traditional inspection process conducted while the facility is being constructed. The ITAAC are not intended to supersede or substitute for existing regulations and inspections, but instead will provide a complimentary process by which to verify that the facility will conform with the certified design. Thus, all requirements such as those found in Quality Assurance Programs derived from Part 50, Appendix B, will still be in effect. Inspection activities will be coordinated in a manner similar to that conducted in the readiness review pilot program for Vogtle.

Summary

The designers are developing the first ITAAC for the lead designs being considered for standardized design certification under 10 CFR Part 52. The ITAAC provide the basis to confirm that key aspects of a facility licensed under Part 52 are built and will operate in accordance with the Atomic Energy Act and the Commission's regulations. Since the ITAAC implement many issues associated with standardized reactor designs, they will play a significant role in the licensing process of the next generation of nuclear power plants in the United States.

Decommissioning Regulations for Power Reactors

Richard F. Dudley, PDNP

Introduction

On June 27, 1988, the NRC issued a package of revised and new regulations to ensure the safe and effective decommissioning of nuclear facilities. These regulations became effective 30 days later. However, the NRC gave licensees until July 26, 1990, to submit reports indicating the manner in which they will comply with the requirements for ensuring funding for decommissioning. When the NRC issued these final decommissioning regulations in 1988, it concluded 10 years of performing technical, environmental, policy, and legal analyses. The NRC also included in the rule its review and responses to public comments received on proposed regulations published in February 1985. Although these regulations apply to all NRC-regulated nuclear facilities, this article will only address their effects on nuclear power reactors.

Acceptable Decommissioning Alternatives

The regulations define "decommissioning" as safely removing a facility from service followed by reducing residual radioactivity to a level that permits the release of the property for unrestricted use. They do not require the removal of non-radioactive structures or structures that have been decontaminated to levels acceptable for unrestricted use. The regulations specify that decommissioning must be accomplished within 60 years of the time when the plant is shut down, although a longer period may be allowed under certain circumstances if necessary to protect the public health and safety. The NRC conducted technical analysis to support the rule and concluded that three basic methods of decommissioning were acceptable:

DECON:	Promptly decontaminate and dismantle the facility.
SAFSTOR:	Place the facility in an isolated, safe storage condition allowing radioactivity to decay before dismantlement.
ENTOMB:	Entomb the facility by encasing radioactive materials in structurally long-lived concrete or other material and store until decay allows release for unrestricted access.

The time limitations for completing decommissioning may not allow for the ENTOMB alternative when long-lived radionuclides are present. A power reactor which has operated for 40 years will likely have inventories of nickel-59, nickel-63, and niobium-94 (half lives of 80,000 years, 92 years, and 20,000 years, respectively) inside the reactor vessel, which would normally preclude using the ENTOMB alternative.

Financial Assurance

The NRC required each power reactor licensee to submit a report by July 26, 1990, to indicate the manner in which it would ensure that funds would be available to decommission the facility. Acceptable funding methods allowed by the regulations include making a prepayment, establishing an external sinking fund, or obtaining a surety method (bond, letter of credit, insurance, or other guarantee). Nearly all power reactor licensees have chosen to establish external sinking funds into which periodic deposits are made so that upon expiration of the facility license, sufficient funds will be available to pay all decommissioning costs.

The regulations specify a minimum amount of funding that must be ensured [$105M for large PWRs and $135M for large BWRs (1986 dollars)] and also provide a formula for annually adjusting these amounts to account for the escalating costs of labor, energy, and waste disposal. The regulations also allow the licensee to use a site-specific cost estimate if it is not less than the minimum value. These decommissioning costs do not include the costs necessary to store and dispose of spent fuel.

Planning for Decommissioning

The regulations in 10 CFR Part 50.75 require each licensee to submit a preliminary decommissioning plan on or about 5 years before the projected end of plant operation. The preliminary plan is to include a site-specific cost estimate, the decommissioning alternative anticipated to be used, major technical actions necessary to decommission the facility, the current situation with regard to disposal of high-level and low-level radioactive waste, the criteria for residual radioactivity, and any other site-specific factors that could affect planning and cost. If necessary, this submittal must also include plans for adjusting the level of funds ensured to be available for decommissioning. Although the plan must contain sufficient detail to justify the site-specific cost estimate, the preliminary decommissioning plans need not be approved by the NRC.

In 10 CFR Part 50.82, "Termination of License," the NRC requires the licensee to submit an application to surrender the license within 2 years of shutdown and not later than 1 year before the operating license expires. The application to surrender the license must be accompanied by or preceded by a proposed decommissioning plan. This plan must include the decommissioning alternative selected, a description of the controls to be used to protect the health and safety of the public, a description of the final radiation survey, an updated cost estimate and a comparison with the amount of funds already collected, and a description of the technical specifications, quality assurance provisions, and physical security plan provisions proposed for decommissioning. After the NRC reviews and approves this decommissioning plan, the staff will issue a decommissioning order which amends the license to establish the specific regulatory requirements for decommissioning at the particular facility.

Radiological Release Criteria

The regulations do not include radiological release criteria since final criteria are to be developed by the Environmental Protection Agency (EPA). Until the EPA develops these criteria, the NRC will continue to use the surface contamination limits in Regulatory Guide 1.86, "Termination of Operating Licenses," 1974. Since 1981, for Part 50 reactor facilities, the NRC has also used as the limit for fixed contamination by gamma-emitting radionuclides, a value of 5 micro-Roentgen per hour above background at a distance of 1 meter. Occasionally, the NRC permits licensees to exceed the 5 micro-Roentgen per hour limit if it can be shown using reasonable occupancy assumptions that the maximum dose commitment to an individual would not exceed 10 mrem per year. Recently, in a June 28, 1991, staff requirements memorandum, the Commission directed the staff of both NRR and NMSS to continue to use these existing criteria (which were in place just prior to the July 3, 1990, publication of the Below Regulatory Concern Policy Statement), in making licensing decisions involving decommissioning.

Prematurely Shutdown Plants

The decommissioning regulations described herein do not specifically address plants that prematurely shut down before their operating licenses expire. The NRC found several concerns after the Fort St. Vrain, Rancho Seco, and Shoreham plants all shut down prematurely and entered the decommissioning process. The most significant of these concerns were (1) the nature and extent of the NRC review under the National Environmental Policy Act (NEPA), specifically whether the environmental impacts of replacement power alternatives must be compared with the environmental impacts of returning a shutdown plant to power operation, (2) the time period for accumulating the funds for decommissioning, and (3) the requirement to pay annual fees (10 CFR Part 171). Over the past several years, the staff has worked with these licensees and with the Commission to resolve all significant uncertainties in these areas. The staff is now conducting decommissioning reviews for these prematurely shutdown plants and has nearly completed the process to approve the decommissioning plan for Shoreham.

Additional Rulemaking Needed

The staff's recent licensing efforts on plants in the decommissioning process have shown that when many existing NRC regulations were promulgated, their applicability to decommissioning was not always evaluated by the staff. Thus, the staff has issued numerous exemptions since strict compliance with certain regulations is not necessary to ensure adequate safety at decommissioning plants. NRR has recently requested that the Office of Nuclear Regulatory Research initiate a rulemaking to consider modifying and clarifying requirements for decommissioning plants in the areas of 10 CFR Part 50.59 applicability, possession-only license issuance, emergency preparedness, security and safeguards, property damage liability insurance, requirements for operator training and requalification and for simulators, fitness-for-duty, 10 CFR Part 50 Appendix J leak testing, maintenance, and fire protection.

U.S. Program For Advanced Liquid Metal Reactor (ALMR) Development

Stephen P. Sands, PDAR, and
Geoffrey R. Golub, PDAR

Within the framework of the Advanced Liquid Metal Reactor (ALMR) Program, the U.S. Department of Energy (DOE) selected the Power Reactor Innovative Small Module (PRISM) design as the liquid metal reactor design to sponsor for NRC design certification. The conceptual PRISM design was developed by General Electric (GE) Company in conjunction with an industrial team comprising Bechtel Power Corporation, Borg-Warner Corporation, Foster Wheeler Corporation, and United Engineers and Constructors, Inc. Research and development support is being supplied by the Argonne National Laboratory, Energy Technology Engineering Center, Hanford Engineering Development Laboratory, and Oak Ridge National Laboratory. In addition, a steering group of utility representatives was involved in the PRISM design effort.

DOE chose to sponsor the PRISM design as part of its National Energy Strategy because of the design's potential for enhanced safety through the use of passive safety systems and greater safety margins, reduced cost through modular design and construction, and possible future contribution to high-level waste management through development of an actinide recycling capability. Although this last alternative has not yet been proposed in the current application, actinides separated from light-water reactor spent fuel could be burned in an ALMR fast-flux core. This process offers the potential to reduce the high-level-waste (HLW) storage requirements for light-water-reactor spent fuel from thousands to hundreds of years.

The NRC is reviewing the PRISM conceptual design and will issue a final Preapplication Safety Evaluation Report (PSER). In order to obtain NRC approval of its prototype, DOE plans to submit a Preliminary Safety Assessment Report (PSAR) in 1995 and a Final Safety Assessment Report in 1997. DOE also plans to apply for standard design certification in 2003 after a prototype demonstration. These plans are based on the current DOE goals to demonstrate the commercial potential for the ALMR by 2010, as called for in the National Energy Strategy.

The PRISM Plant Design

The PRISM plant design consists of three separate power blocks each made up of three reactor modules (Figure 1). Each module has a thermal output of 471 MWt and an electric output of 155 MWe for a total (plant) output of 1395 MWe. Options for one or two power blocks are also possible. The PRISM design contains three turbines, each supplied from a power block. The nuclear steam supply system (NSSS) for PRISM consists of the primary sodium loop and the secondary (intermediate) sodium loop, which receives heat from the primary system and transfers it to the steam generator. The steam generator is the interface for sodium and water systems.

Reactor Module

The reactor module consists of the containment system, the reactor vessel, the core, and the reactor's internal components. The reactor vessel encloses and supports the core, the primary sodium coolant system, and the internal components. The vessel is located just inside the containment vessel, which is located below grade in the reactor silo. The reactor vessel is made of 2-inch-thick, type 316 stainless steel. The reactor vessel is penetrated only in the closure head. It is supported by the floor structure, and the floor structure is supported by seismic isolator bearings to reduce horizontal movement during seismic events.

The reactor core is supported by a beam structure at the bottom of the reactor vessel. The reactor vessel also contains support for storing up to 30 spent fuel and blanket assemblies. The upper head of the reactor vessel is the closure head, consisting of 12-inch-thick, type 304 stainless steel. It would assist in mitigating the effects of hypothetical core disruptive accidents (HCDA). The closure head also supports the intermediate heat exchangers (IHXs) and the electromagnetic (EM) pumps. The reactor vessel is about 62 feet high and just under 20 feet in diameter.

Nuclear Steam Supply System (NSSS)

The main components of the NSSS in PRISM are the reactor module, primary sodium loop, EM pumps, IHX, intermediate sodium loop and steam generators (SGs).

The sodium in the primary loop circulates from the core outlet to the shell side of the IHX to the EM pump and then to the core inlet. This primary sodium loop is contained completely within the reactor vessel, which is hermetically sealed to prevent leakage of the primary coolant. The high saturation temperature of sodium allows large margins to voiding and low system pressure during normal operation. The EM pumps provide the primary sodium circulation. Conventional pump coastdown is not possible because the EM pumps have no moving parts. However, a synchronous machine provides flow coastdown through the EM pumps. Flow coastdown is very important for preventing sodium voiding during a loss of power without reactor scram.

Reactor-generated heat in the primary loop is transferred through the IHX to the intermediate heat transfer system (IHTS). IHTS sodium is circulated by a centrifugal pump. The IHTS operates at a higher pressure than the primary loop so that, if the IHX breaks, the sodium would not flow out of the reactor vessel. The IHTS transfers heat to the SG system, which provides saturated steam at 965 psi to the turbines. A sodium-water reaction protection system mitigates the effects of reactions between sodium and water in the SG.

Core and Fuel

The reference fuel for the ALMR is a uranium-plutonium-zirconium (U-Pu-Zr) alloy. The ferritic alloy HT9 is used for cladding and channels to minimize swelling caused by long burnups. The PRISM core is a heterogeneous arrangement of driver fuel and blankets. The PRISM design has six control rods. The refueling schedule for PRISM calls for replacing one-third of the core every 18 months. The fuel designers cite negative reactivity feedbacks, better heat transfer properties, and competitive costs as advantages of metal fuel. However, a drawback is that the metal fuel has a considerably lower melting point than oxide fuel. Also, the neutronic design employed to achieve the negative feedbacks necessary for a highly desirable passive shutdown feature results in an undesirable positive void coefficient. However, the negative temperature coefficient and operation at temperatures well below sodium boiling are intended to make core voiding a highly unlikely event. The Argonne National Laboratory (ANL) is continuing to develop the metal fuel as part of the Integral Fast Reactor (IFR) Fuel Program for the PRISM design.

Reactivity Control and Shutdown Systems

There are six control rods in the main reactivity control and shutdown system. Inserting any one of the six will shut the core down. The control rods can be inserted using (1) the plant control system (PCS), for normal insertion, (2) the safety-grade reactor protection system (RPS) for rapid insertion, and (3) gravity drop into the core. If this system fails, the operator can send boron balls into the central location of the core which causes shutdown independently of the control rods. The PRISM design also includes passive mechanisms for controlling reactivity: three gas expansion modules (GEMs) consisting of tubes, closed at the top and open at the bottom, filled with helium. If the pumps are running, the static pressure is high, causing the sodium level to rise to a high point in the GEM. However, with the pumps off, the static pressure and sodium level drop, which increases neutron leakage. The reactivity change from the GEMs between these two states is about 70 cents.

Residual Heat Removal

Normal cooling through the non-safety-grade condenser is used for residual heat removal (RHR). If the condenser becomes unavailable, the safety-grade reactor vessel auxiliary cooling system (RVACS) is used for RHR. The RVACS operates through the direct natural circulation air cooling of the containment vessel. The design-basis accident involves a loss of all RHR except the RVACS. Analysis has shown that the RVACS' heat removal rate is sufficient to maintain the temperatures of the internal structures under American Society of Mechanical Engineers (ASME) Level C conditions (1200xF). The PRISM design also contains the non-safety-grade auxiliary cooling system (ACS) to assist the RVACS. The ACS uses natural circulation of the steam generator to remove heat indirectly from the reactor vessel. The ACS can be used in combination with the RVACS to reduce the cooldown time.

The frequency, magnitude, and duration of high temperatures reached during RVACS accident scenarios may be of concern to safety. Therefore, researchers will examine the effect of these temperatures on the life of internal components in the PRISM prototype reactor. However, the simplicity of the RVACS will make it difficult to defeat. The RVACS has two major classes of failure modes: the degradation of RVACS surfaces to lower heat transfer capabilities and the blockage of RVACS flow passages. For flow blockage scenarios, the duration of the flow blockage will determine the outcome of these events.

Containment

The containment is unconventional, consisting of the containment vessel (guard vessel) and a containment dome above the reactor closure head. The containment system is a low-leakage, pressure-retaining boundary separate from the reactor vessel. The containment vessel is located just outboard of the reactor vessel and has no penetrations. A break in the reactor vessel wall will allow sodium to spill into the containment vessel, but the core will still remain covered. The atmosphere in the reactor vessel above the sodium pool is helium. However, the atmosphere in the containment dome is normal air. The designer performed a preliminary analysis to determine radioactive releases for HCDAs with a simultaneous break in the reactor vessel. However, the source term for the PRISM design remains an open issue.

NRC Activities

On July 8, 1986, the Commission published "Regulation of Advanced Nuclear Power Plants, Statement of Policy" (51 FR 24643), and in June 1988, the NRC issued "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants." The Commission encouraged the NRC staff to interact early with designers of advanced reactors to establish licensing guidance that applied to such designs. In accord with the policy statement, the staff would review conceptual designs before receiving any formal application for a construction permit or standard plant review and certification.

In November 1986, the NRC received the PRISM Preliminary Safety Information Document (PSID) for this preapplication review. The Office of Nuclear Regulatory Research (RES) reviewed the design in accordance with the guidance of NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," and a draft Preapplication Safety Evaluation Report (PSER) was issued to DOE in September 1989 after it was reviewed by the Advisory Committee on Reactor Safeguards (ACRS), although the Commission did not give its formal approval of the draft PSER. The NRC staff and the ACRS concluded that the PRISM design provided several features for making a nuclear power plant safer and that design and development should be continued. The draft PSER also identified a number of concerns and open issues that needed to be resolved before the design could be approved.

In May 1990, DOE submitted Amendments 12 and 13 to the PSID to address the issues and concerns identified in the draft PSER. Major design changes were made, including the following:

- adding a low-leakage, pressure-retaining containment dome above the vessel head
- strengthening the core support and vessel head penetration structures to better contain postulated severe core-disruptive events
- adding a diverse reactivity shutdown system, using boron balls at the center of the core
- demonstrating a seismic capability of 0.5g peak ground acceleration above the design-basis ground acceleration of 0.3g adding three gas expansion modules (GEMs) at the core periphery for increased reactivity margin during loss-of-flow events

The PRISM design is currently one of four preapplication reviews being conducted by the Advanced Reactors Project Directorate in NRR. The objectives of the preapplication reviews are identified in SECY-91-202 as follows:

- Identify major issues that could require the Commission to provide policy guidance before the staff initiates actions.
- Identify major technical issues that the staff could resolve in the context of existing regulations and Commission policy, and for which additional Commission guidance is not considered necessary.
- Identify research and development that is needed to resolve noted issues.

Key technical and policy issues being considered during the preapplication review include the following:

- **Metallic Fuel Performance.** The proposed reference fuel design is a U-Pu-Zr metallic fuel with steel alloy HT9 cladding. The staff needs more information on the phase change at high temperature, the fuel extrusion during a significant overpower event, and the eutectic formation and interaction with the cladding. ANL is managing the DOE fuel research program which is still in the development stages and requires close monitoring.
- **Positive Void Coefficient.** The proposed core design has a positive void coefficient that could result in a large positive reactivity addition should sodium boiling occur in the center of the core. Redundant and diverse coolant flow pumps and GEMs were designed to deal with this problem. The NRC staff is evaluating the preventive and mitigative features of the PRISM design to assess the safety significance of this issue.
- **Hypothetical Core Disruptive Accident (HCDA).** Potential HCDA initiators include a large unprotected reactivity insertion from core voiding. The proposed reactor vessel and containment are each designed to

contain the postulated HCDA energy without breaching. The staff is evaluating the magnitude of the mechanical energy released from an HCDA and assessing the integrity of the primary system, containment, reactor vessel, and vessel head.

- **Source Term.** The preapplicant has proposed using a source term unlike that used for current light-water reactors to account for the liquid metal reactor design. Release estimates from the metallic fuel and decontamination factors for the sodium pool and cover gas under the containment dome need to be supported by experimental data.

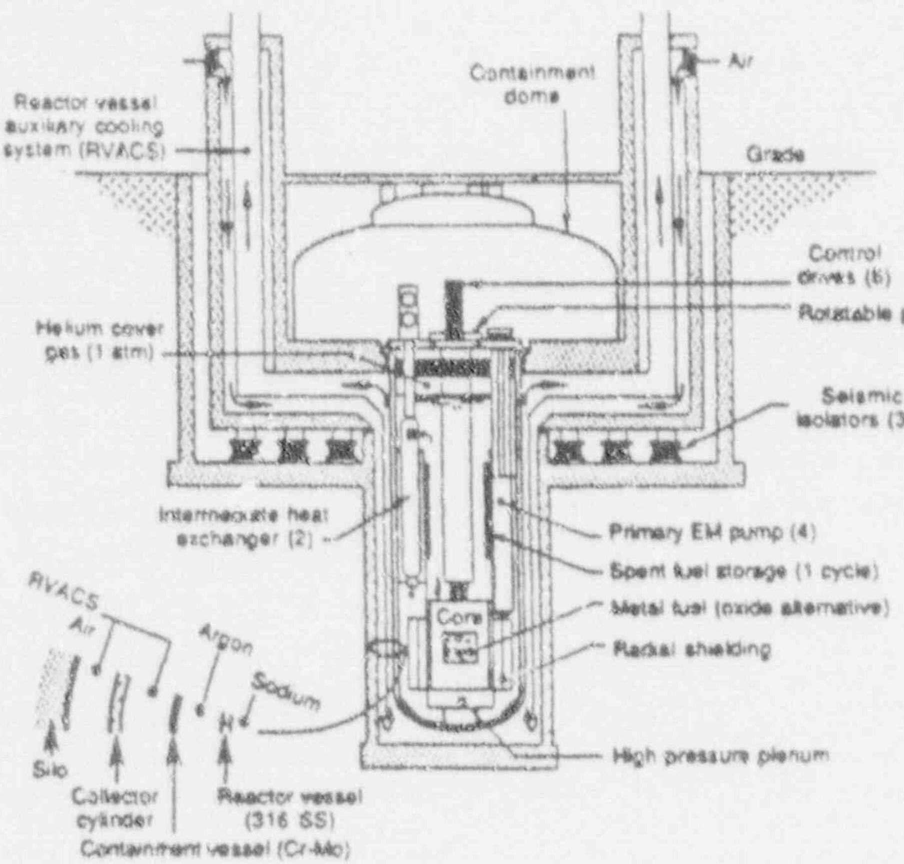
- **Operator Staffing and Control Room Design.** The proposed control room utilizes advanced instrumentation and control systems to support nine reactor modules from one room with as few as five licensed operators. The staff needs to evaluate whether the proposed design and staffing can effectively manage multiple plants simultaneously during accident, transient, and shutdown situations.

- **Passive Residual Heat Removal System.** The proposed design utilizes natural convective air flow around the lower containment vessel through a hot air riser space. The system is completely passive and in continuous operation, providing increased reliability. However, this system alone could leave the plant at elevated reactor temperatures for long periods after accidents. The staff is evaluating the acceptability of highly reliable, active, non-safety-grade, heat-removal systems to reduce the frequency, magnitude, and duration of challenges to the passive system.

- **Seismic Isolation.** The proposed design uses large seismic isolators to support the nuclear island to reduce the magnitude of horizontal ground acceleration transmitted to the safety-grade nuclear island structures, systems, and components. The staff will continue to evaluate the isolator system as additional design information and test results become available.

- **Emergency Planning.** The preapplicant has postulated a very low probability of exceeding the Protective Action Guideline lower limits at the site boundary and has proposed reduced requirements for offsite emergency planning. This is a policy issue that the staff will raise to the Commission for further guidance before evaluating emergency planning for the PRISM design.

The staff is reviewing the new design and evaluating key issues and will incorporate the results of its evaluation in a final PSER expected to be completed in November 1992.



REACTOR MODULE FOR PRISM

Enhancing Safety Using PRA and IPE Techniques and Results

by Dennis F. Kirsch, Regi[illegible]n V

Introduction

The nuclear utility industry is performing individual plant examinations (IPEs) using probabilistic risk assessment (PRA) techniques, to quantify plant risk resulting from sequences that can initiate several severe accident events. PRA techniques enable plant management to determine how existing and proposed equipment and the procedures and practices for operation and maintenance affect the overall plant risk. The licensee can use the results of the IPE process to assess the effects on operational safety of activities in several broad areas. For example, the IPE results can be applied to qualitatively assess the effects that scheduled maintenance and testing have on average plant risk, assess the effects of design modifications on plant risk, and optimize the technical specifications to minimize plant risk.

The results of calculations indicate that, while plant safety systems are maintained or tested online, the plant's risk of core damage increases in a manner directly proportional to the amount of time the system is unavailable. For this reason, the benefits of voluntary online tests and maintenance should be weighed against the risk incurred.

Outage Safety Assessments

While IPEs and most other PRAs have only considered plant configurations typical of normal, full-power operation, PRA techniques can be adapted to provide insights about risk for configuration other than full power. Initial work in this area has been summarized in "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," NUREG-1449 (DRAFT REPORT), February 1992. During outages, the licensee can utilize PRA techniques to identify and qualitatively address accident scenarios that are of high relative probability and can result in severe consequences. For example, to evaluate the special risk considerations associated with PWR mid-loop operation, the licensee can use PRA techniques to qualitatively identify and evaluate the relative effect on total risk resulting from removing equipment or systems from service during mid-loop operation or from entering mid-loop operation with certain combinations of equipment unavailable. The licensee should carefully evaluate the advantages and disadvantages of performing voluntary online tests and maintenance of components or systems that can affect risk during mid-loop operation.

To perform realistic outage safety assessments, the licensee can use risk-based failure modes and effects analysis (FMEA) of shutdown conditions and the effects of special plant conditions on the risks attendant with those conditions. RISK-BASED TECHNICAL SPECIFICATIONS

To minimize the risk of potential accident sequences at the plants, PRA techniques and IPE results can be used to design the allowed outage times and surveillance time intervals in the technical specifications to reduce core damage frequencies. This application of risk assessment would provide a logical process by which to balance plant safety with the need for plant safety systems to be reliable and operable and the need to periodically test and maintain the systems. PRA techniques have been used in revising technical specifications for actuation instrumentation and the staff expects other similar applications to increase in the future.

Applicability To Daily Operational Decisions

The use of the IPE methodology and results could contribute significantly to the daily planning and scheduling of test and maintenance activities.

If all systems are in service and are operable, the risk, or core damage frequency (CDF) (events per year), is minimal. However, risk increases when equipment and systems must be removed from operability status during plant operation for test and maintenance activities. The actual magnitude by which risk increases depends upon the relative importance of the component or system in the event tree and is directly proportional to the out-of-service time. Risk also increases due to multiple component or system inoperabilities. These factors cause risk to vary continuously throughout the operating cycle.

To determine if the removal of a safety related component increases risk unacceptably, the licensee should establish an upper limit risk guideline. However most licensees have not yet done this.

PRAs can be modified to reflect actual plant equipment configuration used to determine if or when to remove equipment from service for testing and maintenance. For example, certain surveillance testing, required by technical specifications, could be timed so that the necessary system outage contributes as little as possible to the total risk in combination with all other equipment in test or maintenance status. The more progressive utilities are considering a methodology for assessing the advisability of scheduling and performing needed test and maintenance activities which considers the status of all safety systems.

In scheduling surveillance testing and maintenance on safety-related systems to take place concurrently (for example, during the day shift when the largest number of maintenance personnel are onsite), the licensee should consider the effects of those activities on the ability of the plant to mitigate events. Even though a strict interpretation of technical specifications may allow such a practice, this is not a conservative operating philosophy. Most utilities understand the risks of that kind of approach. However, the industry has not fully developed an effective methodology for scheduling test and maintenance activities to minimize risk, or maintain risk within recommended guidelines. In preventive maintenance, recent interpretation of the Inspection Manual, addressed voluntary entry into limiting conditions of operation (LCOs) to perform preventive maintenance. According to this interpretation, preventive maintenance (PM) procedures on equipment that is performed online should improve safety (that is, reduce risk or improve reliability), and should not be performed as a matter of convenience. This interpretation implies that unenlightened conformance to PM program scheduling and requirements is short-sighted.

Plant Experience

Region V has recently examined the industry's experience that correlates the effect of system outage time (resulting from online test or maintenance) to the increase in risk (expressed as events per reactor year). Region V examined the three safety systems for which utilities report unavailability data to the Institute of Nuclear Power Operations (INPO): high pressure safety injection, emergency diesel generation, and auxiliary feedwater. The region used a baseline CDF for the system, assuming no unavailability. The region compared the baseline CDF to CDFs for a 1-percent system unavailability and the system unavailability as reported to INPO.

The study concluded the following:

a. The licensees for plants examined had not calculated the risk for core damage based upon the actual number of hours of unavailability. Accordingly, management was not aware of the true integrated risk.

b. Each plant has one system, or component, that contributes the most to CDF increase for each 1 percent of unavailability.

c. The magnitude of a CDF increase for any system, assuming a 1-percent unavailability, varies widely among plants (from a fraction of a percent to about 20 percent).

d. The magnitude of CDF increases throughout the year for each plant with the number of hours the safety system is unavailable.

Conclusion

The industry is establishing a technically defensible methodology to minimize risk, improve reliability, and improve safety in the selection and implementation of surveillance testing, corrective maintenance, and preventive maintenance activities. Much work remains to be done in this area. This effort can greatly improve overall plant safety.

IPE methodology and results can be applied to a wide range of licensee organizations and activities. The challenge for the nuclear industry is to devise technically defensible mechanisms to use IPE results to effect real and measurable improvements in plant safety.

☆U.S. Government Printing Office : 1994 — 312-473/40010