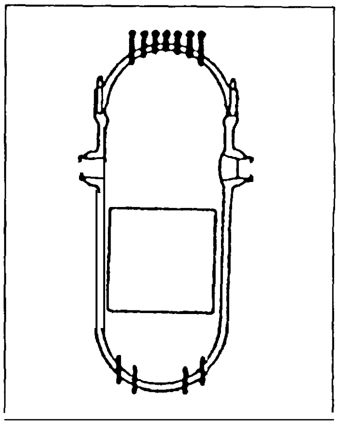
Safety of Aging Nuclear Plants 2

Unchecked, aging degradation has the potential to reduce the safety of operating nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC), the commercial nuclear power industry, and others engage in a range of activities addressing the challenges imposed by power plant aging. Many aging mechanisms are plant-specific and extensive research efforts have been developed to address them, but no technically insurmountable industry-wide safety obstacles have been identified.

This chapter examines the safety issues related to nuclear power plants as they age. The frost section describes the causes and effects of aging degradation on nuclear power plant systems, structures, and components. The second section reviews the institutions involved and their roles in assuring the safety of aging nuclear power plants. The third section describes industry and regulatory processes used to address the safety impacts of plant aging. The fourth section discusses the public and occupational health and safety goals established in current policy as they relate to aging nuclear power plants.



THE CAUSES AND EFFECTS OF NUCLEAR POWER PLANT AGING

As defined by the NRC, aging is "the cumulative, time-dependent degradation of a system, structure, or component (SSC) in a nuclear power plant that, if unmitigated, could compromise continuing safe operation of the plant. ' The nuclear power industry takes a broader view, noting that unmitigated aging degradation can impair the ability of any SSC to perform its design function,[2] possibly affecting not only safety, but also the economic performance and value of a plant.

[1] U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) Program Plan*, NUREG-1 144, Rev. 2 (Washington DC: June 1991).

[2] MPR Associates and the Electric Power Research Institute, *Nuclear Power Plant Common Aging Terminology,* EPRI TR-1OO844 (Palo Alto, CA: Electric Power Research Institute, November 1992), p. C-1.

Many nuclear power plant SSCs are subject to aging degradation, which can cause a variety of changes in the physical properties of metals, concrete, electrical cables, and other materials. These materials may undergo changes in their dimensions, ductility, fatigue capacity, or mechanical or dielectric strength. Aging degradation results from a variety of physical or chemical processes such as corrosion, fatigue, fabrication defects, embrittlement, and mechanical effects (box 2-A). These aging mechanisms can act on power plant components from high heat and pressure, radiation, and reactive chemicals. Some plant operating procedures such as changing power output and even equipment testing also create stress for plant components.

Absent effective management, aging degradation increases the probability that any SSC will fail to function properly. A failure may initiate a system transient or accident sequence, and so

Box 2-A-Metal Aging Degradation Mechanisms

This is a partial listing of aging degradation mechanisms for metals, with examples of effects greater than anticipated in plant design and methods used to address them.

Corrosion is the deterioration of a material resulting from reactions with its environment. Some steam generator components, piping, pressure vessel internals, and other plant areas have experienced more extensive corrosion than originally assumed during plant design. Major forms of corrosion include wastage, stress corrosion cracking, erosion/corrosion, crevice corrosion, and intergranular attack. Methods of addressing corrosion for existing components have been developed, including inspections for signs of deterioration, control of water chemistry, or replacement with resistant materials or designs.

Fatigue is the deterioration of a material from the repeated cycles of thermal or mechanical loads or strains. The number of cycles a material will tolerate until failure is used to classify it as either low cycle (withstanding less than 10 or 10 cycles) or high cycle. Together with the number of cycles expected, the magnitude of expected cyclic loads is a key design condition. Some fatigue failures in piping and other components have occurred, often resulting from larger than anticipated loads or combinations with other degradation mechanisms (e.g., corrosion). Methods of addressing fatigue for an existing component include inspections and more accurate estimates and monitoring of the magnitude and frequency of cyclic loads.

Fabrication defects can contribute to more rapid fatigue cracking and corrosion. Casting and forming defects and weld-related defects embedded in a material may worsen from cyclic loadings, or such defects may become exposed by corrosion. The distribution of flaws in a material is a key consideration, and design codes specify the acceptable level of fabrication defects. Methods of addressing fabrication defects for an existing component include inspections using nondestructive examination techniques to detect embedded flaws early, and repairs when necessary.

Embrittlement is a change in a material's mechanical properties such as decreased ductility and reduced tolerance to cracks resulting from thermal aging or irradiation. Some embrittlement has been found to be more rapid than originally anticipated in plant design. Neutron irradiation of reactor pressure vessels (RPVs), for example, can lead to a more rapid loss of ductility than expected, particularly when copper and nickel are contained in RPV weld materials. Methods of addressing embrittlement for an existing component include more accurate estimates of thermal exposure and neutron fluence histories and their effects, revised operations (e.g., arranging fuel to reduce neutron flux to certain RPV regions), and component replacement or refurbishment (e.g., RPV annealing).

Mechanical effects include vibration, water hammer, and wear. Vibration and water hammer can result from fluid flows and result in loads greater than explicitly considered during design, contributing to fatigue failures and damage to pipes, valves, and pumps.

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[1] Structural Integrity Associates, Inc., *Component Life Estimation: LWR Structural Materials Degradation Mechanisms*, EPRI NP-5461 (Palo Alto, CA: Electric Power Research Institute, September 1987).

Box 2-B–Reactor Pressure Vessel Embrittlement

After years of neutron bombardment from the reactor core, the steel that comprises a reactor pressure vessel (RPV) can gradually lose some of its toughness in **a process called embrittlement. Neutron embrittlement is exacerbated if the steel or weld materials contain trace amounts of copper or nickel. The greatest potential problem of RPV embrittlement is the threat of pressurized thermal shock (PTS). PTS leading to RPV cracking may occur during certain abnormal plant events when relatively cool water is introduced into a reactor vessel while under high pressure after a loss of coolant accident.** U.S. **Nuclear Regulatory Commission** (NRC) requirements and the American Society of Mechanical Engineers (ASME) Code for inspection and analysis are designed to ensure that the pressure vessels are tough enough to resist cracking if PTS occurs.[1]

Although the role of copper and nickel in RPV embrittlement has been known for the past two decades, several older plants were constructed using weld materials with traces of those metals. Because of the original conservative

[1] **10 CFR 50.60 *et seq.*; U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988; 10 CFR 50 Appendices A, G and H; and ASME Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."**

(Continued on next page)

become noticeable immediately. However, not all SSC failures are readily observable. For example, the failure of an emergency diesel generator (EDG), which is not used during normal operations but is needed only for backup power if offsite power is lost, may not be noticed until it is tested or called on to supply power. Also, accidents may induce some SSC failures. For example, aging may render electrical equipment vulnerable to the conditions that arise from an accident, such as changes in humidity, chemical exposure, radiation, and temperature.[3]

The basic processes of nuclear power plant aging are generally, if imperfectly, understood; operating experience and research provide ongoing improvements in the scientific understanding and ability to predict and address aging effects. There is a fairly limited set of degradation mechanisms, a large commonality in materials used, and fairly similar plant operating conditions. However, due to the diversity in power plant designs, construction and materials used, operating conditions and histories, and maintenance practices, the specific effects of aging, although similar, are unique to each plant. Even near-twin units with the same management at the same site can have substantial differences in the remaining lives of their major SSCs.

For example, consider Baltimore Gas and Electric's two 825-megawatt (MWe) Calvert Cliffs units. Construction licenses for both units were issued in July 1969, and the same principal contractor was responsible for both units. The second unit was completed only 2 years after the first and has a reactor pressure vessel (RPV) free of copper and nickel, making it relatively immune to neutron embrittlement (box 2-B). The Unit 1 RPV, however, was built before the discovery that neutron embrittlement can occur more rapidly in steels with trace amounts of copper and nickel. As a result, special procedures and mitigation measures are necessary for Unit 1 to attain its full licensed life.[4]

[3] Electrical equipment required to perform a safety function during or following a design basis event must be qualified in accordance with 10 CFR 50.49, which includes aging considerations. As discussed below, the NRC and the commercial nuclear power industry are examining the adequacy of these requirements.

[4] Barth Doroshuk, Principal Engineer, Nuclear Engineering Department, Baltimore Gas and Electric Co., personal communication June 9, 1992.

Box 2-B–Reactor Pressure Vessel Embrittlement--(Continued)

engineering designs and relative youth of most plants, only one plant to date, Yankee Rowe, has faced early retirement for embrittlement-related concerns. Fifteen other operating units currently do not meet generic screening limits set by the NRC, and another two will similarly not satisfy the generic guidelines before the end of their licensed lives.[2] However, the NRC's generic screening limits are intentionally conservative and do not necessarily indicate an unacceptable level of embrittlement. Rather, failing to meet the genetic limit indicates the need for a more detailed (e.g., plant-specific) analysis based on the ASME Code. During 1993, the NRC plans to validate licensees' plant-specific data and analyses to determine that current requirements are met.[3] While the NRC's preliminary assessment is that the industry RPV analyses are adequate, the differing professional opinions between NRC staff and engineers in the case of Yankee Rowe indicate some potential for a challenging process of resolution.

The NRC and the commercial nuclear power industry both perform extensive research on RPV issues! Improved analytical and nondestructive examination (NDE) methods (e.g., to characterize better the size and distribution of RPV flaws, and the effects of cladding in crack propagation) may help determine if conservatism in currently required margins can be reduced. In a recent report for the Electric Power Research Institute, the ASME Section Xl Task Group recommended updating the current code based on improvements in such technical areas.[5] Several of the recommendations could result in longer estimated lives for RPVs, as more accurate methods replace conservative assumptions in the present code. If more accurate analyses indicate that mitigation is needed, the rate of embrittlement can be reduced by methods such as shielding the RPV wall, or placing the most depleted fuel nearest the RPV's most sensitive areas to reduce the rate of neutron flux. Other options for reducing PTS risks are safety system design and operating procedures that reduce the frequency and severity of challenges (e.g., controlling heat up and cool down rates, reducing pressure prior to emergency coolant injection, and heating or mixing emergency coolant).

To restore the toughness lost to embrittlement, a process called annealing has been routinely used at several nuclear power plants in the former Soviet Union and for U.S. naval reactors.[6] Annealing involves heating a vessel to sufficiently high temperatures to allow the metal to regain its original properties. No such effort has been made for commercial reactors in the United States, although EPRI and the NRC have supported research on the topic.[7] After witnessing and investigating a successful Soviet annealment effort, a U.S. NRC-sponsored team concluded that although there are some technical differences and issues to resolve, the basic process maybe applicable to U.S. vessels.[8]

Embrittlement is not the only aging mechanism that can affect RPVs. Figure 2-1 shows an NRC summary of the key degradation sites, aging causes, failure modes, and maintenance and mitigation actions for pressurized water reactor (PWR) RPVs.[9]

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2 U.S. Nuclear Regulatory Commission, "Status of Reactor Vessel issues Including Compliance with 10 CFR Part 50, Appendices G and H," SECY-93-048, Feb. 25, 1993. NRC also noted that one additional unit with an indefinitely deferred construction schedule would not meet the limit at the end of its licensed life.

3 Ibid.

4 U.S. Nuclear Regulatory Commission, *Proceedings of the Seminar on Assessment of Fracture Prediction Technology: Piping and Pressure Vessels,* NUREG/CP-0037 (Washington, DC: February 1991); and "Pressure Vessel Life-Cycle Management," *EPRI Journal* October/November 1991, pp. 32-33.

5 ASME Section XI Task Group on Reactor Vessel Integrity Requirements, *White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions,* EPRI TR-1OO251 (Palo Alto, CA: Electric Power Research Institute, January 1993).

6 MPR Associates, Inc., *Report on Annealing of the Novovoronezh Unit 3 Reactor Vessel in the USSR,* NUREG/CR-5760 (Washington, DC: U.S. Nuclear Regulatory Commission, July 1991).

7 Oak Ridge Associated Universities, *The Longevity of Nuclear Power Systems,* EPRI NP-4208 (Palo Alto, CA: Electric Power Research Institute, August 1985), Appendix A.

8 MPR Associates, Inc., *Report on Annealing of the Novovoronezh Unit 3 Reactor Vessel in the USSR,* NUREG/CR-5760 (Washington, DC: U.S. Nuclear Regulatory Commission, July 1991).

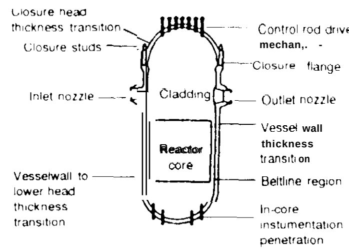
9 U.S. Nuclear Regulatory Commission, *NPAR Program p/an,* NUREG-1144, Rev. 2 (Washington, DC: June 1991), p. 6.24,

Figure 2-I—Nuclear Plant Aging Research Program Summary of Pressurized Water Reactor RPV Aging Issues

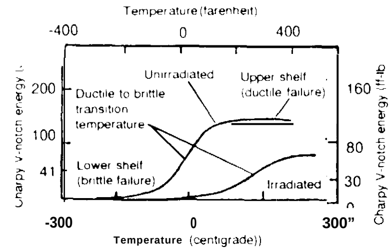
SOURCE: U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) Program Plan,* NUREG-1 144, Rev. 2 (Washington, DC: June 1991, p. 6.24.

Jnderstanding and managing aging in PWR pressure vessels

Materials	Vessels	Low alloy carbon steel SA-533B, SA-508-2, SA 302B
	Cladding	Type 308 SS and 309 SS
	Weldments	Submerged arc (granular flux linde 80, 91, 124 and 1092 manganese-molydenum nickel filler wire) narrow gap submerged arc, sheilded metal arc, and electrostag
	Closure studs	SA-540 Gr. B24 Class 3
Stressors and environment	Neutron flux and fluence, temperature, reactor coolant, cyclic thermal and mechanical loads, preloads and boric acid leakage	



Typical PWR vessel showing *important degradation sites*



Effect of irradiation on the *Charpy impact* energy for a nuclear pressure vessel steel

UNDERSTANDING AGING (materials, stressors, and environmental interaction) | **MANAGING AGING**

SITES	AGING CONCERNS	SERVICE INSPECTION. SURVEILLANCE & MONITORING: NRC requirements	SERVICE INSPECTION. SURVEILLANCE & MONITORING: Recommendations	MITIGATION
Beltline region	Irradiation embrittlement • chemical composition of vessel materials • drop in upper shelf energy (USE) • shift in reference nil-ductility-transition-temperature Environmental fatigue	Surveillance to assess irradiation damage (shift in RT and drop in USE) (10 CFR 50 App. H, Reg. Guide 1.99, Rev 2) Pressurized thermal shock (PTS) screening criteria (10 CFR 50.61) PTS rule, RG 1.154 Damage evaluation (10 CFR 50 App. G) Pressure-Temperature (P-T) limits during heatup, cool-down, criticality, and inservice leakage and hydrostatic pressure test to prevent nonductile fracture (Tech. Spec. requirement) Volumetric examination of all welds during each inspection interval (10 CFR 50 App. G) Low temperature overpressurization (LTOP) protection setpoint (technical specification requirement) Volumetric examination of all welds during each inspection interval (10 CFR 50.55a, IWB-2500, Reg. Guide 1.150, Rev. 1) Flaw evaluation (10 CFR 50.55a, IWB-3000) Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-500)	Include fracture toughness and tensile test specimens in surveillance program Develop use of reconstituted and miniature specimens Develop techniques for in situ determination of mechanical properties Perform accelerated irradiation tests of reconstituted specimens Revise Reg. Guide 1.99 Rev. 2 to account for phosphorus with low copper Use state-of-the-art ultrasonic inspection techniques for *improved reliability of defect detection, sizing, and* characterization • automated amplitude-based systems • *tip diffraction techniques* • large-diameter focused transducer Use fatigue crack growth curves (ASME SC XI, Appendix A) Develop acoustic emission monitoring to detect crack growth (appendix being developed for ASME Section XI)	Neutron flux reduction Inservice annealing (ASTM E 509-86) Determine effects of annealing and re-embrittlement rate
Outlet/inlet nozzles	Environmental fatigue Irradition embrittlement function of nozzle elevation	Volumetric examination of all nozzle to vessel welds during each inspection interval (IWB-2500) Volumetric and surface examination of all dissimilar metal welds during each inspection interval (IWB-2500)	Use online fatigue monitoring (monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures) Evaluate irradiation embrittlement damage	
Instrumentation nozzles CRDM housing nozzles	Environmental fatigue	Visual examination of external weld surface of 25 percent of nozzles during system hydrostatic test (IWB-2500)		
Closure studs	Environmental fatigue • preload cycles during head replacement, boric acid corrosion (if leakage occurs)	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (IWB-2500)		

The useful lives of many power plant components, such as some pumps and valves, are shorter than the expected life of the entire plant. These components are replaced, refurbished, or repaired as part of regular maintenance efforts. In contrast, many other SSCs are designed to last the entire life of a plant. In fact, many of these long-lived SSCs, including most RPVs and concrete structures, appear adequate for periods longer than current license terms. However, some SSCs—such as certain steam generators (box 2-C), RPVs incorporating certain materials, and certain water

Box 2-C-Steam Generator Tube Corrosion and Cracking[1]

Steam generators (SGs) are integral to pressurized water reactors (PWRs), which comprise over two-thirds of U.S. plants. Weighing between 250 to 675 tons, they are large heat exchangers located within a plant's primary containment and within the reactor coolant pressure boundary to transfer energy from the radioactive primary reactor coolant to the nonradioactive secondary steam circuits that turn the turbines. Each PWR has two or more SGs depending on plant design. Although originally designed to last the life of a plant, a variety of mechanisms including corrosion, denting, cracking, and intergranular stress corrosion cracking, have been found to degrade the thousands of tubes in many SGs much more rapidly than expected. Degradation can lead to leaks of radioactive primary coolant and, in extreme oases, ruptured tubes leading to more severe plant problems. Each PWR has a unique SG degradation history due to the diversity of design and materials and conditions such as water chemistry and plant operating history.

Several methods are used to control SG degradation. Improved water chemistry is now widely used to reduce the rate of degradation? inspections using *nondestructive* examination techniques are used to determine the condition of the tubes. When inspections detect unacceptable levels of damage (e.g., cracks greater than 40 percent of a tube's wall thickness), various repair methods are used. Plugging removes a tube from service. An alternative to plugging involves sleeving, or inserting anew tube inside the damaged portion of the original tube. Over 23,000 sleeves had been installed in domestic SGs as of 1990 (84 percent of which were at only four plants). Sleeved tubes remain subject to degradation and may later need plugging. **Heat treatment,** chemical cleaning, and other methods have also been used.

A plant can continue operating with a number of plugged tubes, as specified in plant operating manuals, although plant efficiency is reduced with increasing numbers of sleeved or plugged tubes. When too many tubes degrade too much, continued plant operation at its rated output requires steam generator replacement. Since 1981, steam generators at more than 10 plants have been replaced, and several more are under consideration. Replacement costs are high, often $100 to $200 million, and the work can take several months. For example, Duke Power Co. anticipates spending $600 million on steam generator replacements for its McGuire-1 and -2 and Catawba-1 plants between 1995 and 1997? A group of nine utilities has formed the Steam Generator Replacement Group to make a volume purchase and thus reduce the replacement costs for its 16 PWRs.

The NRC and the commercial nuclear power industry continue working to improve the accuracy and applications of nondestructlve examination techniques for steam generators. The NRC's standard for plugging or repairing a tube is the detection of a crack of a specific length extending through more than 40 percent of the tube. However, the NRC has approved the use of different criteria for a few plants that have microcracks, and the agency continues to investigate **alternate** criteria[4] **Figure 2-2 shows an** NRC summary of the key degradation sites, aging causes, failure modes, and maintenance and mitigation actions for PWR steam generator tubes.[5]

1 Unless otherwise noted, this information is condensed from L. Frank, *Steam Generator Operating Experience, Update for 1989-1990,* NUREG/CR-5796 (Washington, DC: U.S. Nuclear Regulatory Commission, December 1991); and S.E. Kuehn, "A new round of steam generator replacements begins," *Power Engineering,* Juiy 1992, pp. 39-43.

2 PWR Secondary Water Chemistry Guidelines, Rev. 2, EPRi NP-6239, December *1988.*

3 6'Dljke Chooses B&W International to Supply 12 Steam Generators," *Nucleonics Week,* vol. 33, No. 27, July 2, 1992, p. 3.

4 U.S. Nuclear Regulatory Commission, *Voltage-Based Interim Plugging Criteria for Steam Generator Tubes-Task Group Report,* NUREG-1477 draft (Washington, DC: June 1993).

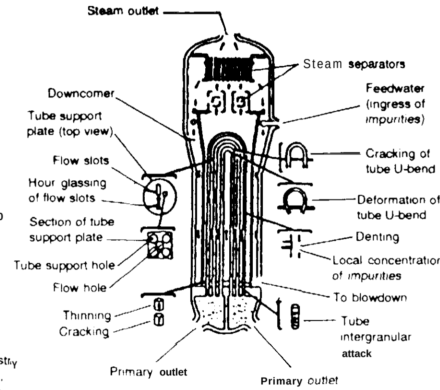
5 U.S. Nuclear Regulatory Commission, *NPARProgram Plan,* NUREG-1 144, Rev. 2 (Washington, DC: June 1991), p. 6.12.

Figure 2-2—Nuclear Plant Aging Research Program Summary of Pressurized Water Reactor Steam Generator Tubes Aging Issues

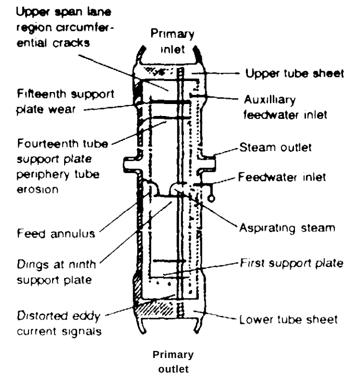
SOURCE: U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) Program Plan,* NUREG-1 144, Rev. 2 (Washington, DC: June 1991), p. 6.30.

Understanding and managing aging of PWR steam generator tubes

Materials	Tubes	Inconel &30 or 690
	Tuba sheet	SA 508 clad with Ni-Cr-Fe alloy (equivalent to SB 16s)
	Tuba supports	SA 2s5 Gr C Ferritic SS Type 405 or 409
	Sleeves	Inconel 625 or nickelbonded on outside surface if Inconel 600 or 690
	Plugs	Inconel 690
Steam generator types	Recirculating Once-through	Westinghouse, Combustion Engineering, Babcock & Wilcox
Stressors and environment	Residual stresses, *primary coolant chemistry (primarily* hydrogen concentration) secondary coolant chemistry (chlorides, oxygen, copper, sulfates), phosphate chemistry resin leakage from condensate polisher, brackish water, temperature, flow-induced vibration, flow-velocities, and operating transients	



Recirculating steam *generator with indicated problem areas*



Once- *Through* steam *generator with indicated problem* areas

UNDERSTANDING AGING
(materials, stressors, and environmental interaction)

TYPES	SITES	AGING CONCERNS
Recirculating inside surface	U bends roll transition and dentedregions	PWSCC (pure water SCC)tubes with low mill annealing temperature are more susceptible
	Tube plugs	
Recirculating outside surface	Ho! lag *tubes in tube-to* tubesheet crevice region	IGSCC IGA
	Cold lag side m sludge pile or where scale containing copper deposits is found	Pitting
	Tubes in tube support regions	Denting
	Inadequately supported tube if dentad near the top support plate	High-cycle fatigue
	Contact points between tube and antivibration bar	Fretting
	Tubes above tubesheet	Wastage
OnC%-through Outslda Surface	Tubas	Erosion-corrosion Fatigue
	Tubes in upper tubesheet region	Environmental fatigue

MANAGING AGING

SERVICE INSPECTION, SURVEILLANCE & MONITORING

NRC Requirements

Volumetric examination of hot leg side, U-bend portion, and (optionally) cold leg side of tubes in recirculating steam generators (IWB-2500)

Volumetric examination of the entire *length of tubing in once-through steam* generators (IWB-2500)

Frequency of inspection and number of tubes to be inspected (minimum of 3 percent of all tubes) are determined by technical specifications (Reg. Guide 1.83)

Standards for allowable flaws in recirculating steam generators (standards for once-through steam generators are being prepared) (IWB-3521)

Flaw acceptance criteria determined by technical specifications (IWB-3630)

Criteria for determining necessity of plugging degraded tubes (Reg. Guide 1.121)

Unscheduled inservice inspection of each steam generator is required when primary to secondary tube leaks exceed the limits defined in technical specifications

Recommendations

Follow Steam Generator Owners Groups' guidelines for continous monitoring and control of secondary water chemistry

Reduce uncertainties in inspection results *and quantify flaw growth rates*

Monitor field performance of various sleeve designs

Perform inservice inspection of tube plugs

MITIGATION

Prevent transient conditions in secondary water chemistry, install filters between condensate polishers and steam generators. Use ultrafiltration of makeup water and remedy condenser leakage as quickly as possible

Use shotpeening and rotopeening to introduce compressive residual stresses on tube inner surface in the roll transition region, and anneal U-bends to reduce PWSCC

Apply nickel plating on the inner surface of the tubes to prevent PWSCC crack initiation and propagation

Use tube rolling to eliminate tube sheet crevices and use *crevice flushing, crevice alkalinityneutralization, alkaline* impurity control, acid chloride elimination, hot soaks, sludge lancing, pressure pulse, water slap, chemical cleaning, and boric acid additions to control IGA/IGSCC

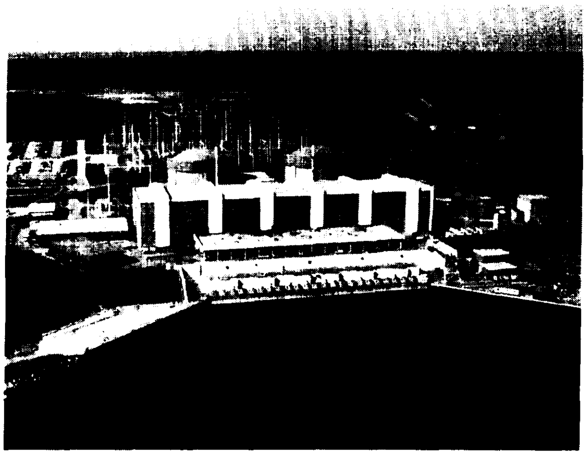
Eliminate copper pickup by use of titanium or stainless steel condenser tubes and replace the copper-bearing alloys in the feedwater train to reduce pitting and denting

Use all-volatile treatment water chemistry, sludge lancing, chemical cleaning, hot soaks, hot blowdown and flushing and elimination of hideout chemical concentration to control wastage

Use chemistry control to prevent concentration of impurities leading to fatigue crack initiation in once-through steam genrators

Use lane-flow blocker in once-through steam generators to mitigate environmental fatigue

PHOTO CREDIT: BALTIMORE GAS AND ELECTRIC CO.



Life-cycle management activities at Baltimore Gas and Electric Co.'s 2 Calvert Cliffs nuclear power plants could be useful in future license renewal efforts.

system piping-may experience more rapid aging degradation than originally anticipated in plant designs. Because few nuclear power plants are in the second half of their 40-year licensed lives, operating experience with the aging of long-lived SSCs remains limited.

INSTITUTIONS FOR ASSURING THE SAFETY OF AGING PLANTS

Under the Atomic Energy Act of 1954 (AEA),[5] as amended, the NRC is responsible for regulating civilian nuclear power facilities 'to assure the common defense and security and to protect the health and safety of the public."[6] To ensure the safety of operating nuclear plants, the NRC performs a variety of activities, including the development and documentation of the "licensing bases' that specify plant design requirements and operation and maintenance (O&M) practices; the inspection and enforcement of license requirements; the performance of technical research and analysis; and the modification of regulatory requirements as needed. All of these activities are involved in addressing power plant aging to assure safe operations.

Although the NRC plays a central role in assuring nuclear power plant safety, the AEA actually assigns the primary responsibility for the safe operation of a commercial nuclear plant with the plant operator, or licensee.[7] Each licensee is ultimately responsible for the design, operation, and maintenance of its plant-not only to meet NRC requirements but to assure safety. To pool resources, share experiences, and coordinate efforts, the U.S. nuclear electric utilities have established several industry-wide organizations concerned with safety and other issues. Notable among them are the Electric Power Research Institute (EPRI), the Institute of Nuclear Power Operations (INPO), and the Nuclear Management and Resources Council (NUMARC).

EPRI was formed in 1973 to perform research and development (R&D) for a broad range of electric utility industry technologies, including nuclear power production. As discussed below, EPRI has sponsored a great deal of R&D directly related to nuclear plant aging issues over the last two decades, ranging from basic material science to improved maintenance practices. The organization helped prepare several of the 10 "industry reports" on license renewal that were eventually submitted to the NRC by NUMARC. Most, but not all, nuclear utilities are EPRI members. As of 1992,7 utilities operating 23 of the Nation's 107

5 Atomic Energy Act of 1954, Public Law 83-703, 68 Stat. 919.

6 These responsibilities were originally granted to the Atomic Energy Commission (AEC). The Energy Reorganization Act of 1974 (Public Law 93-438) transferred these responsibilities from the AEC to the U.S. Nuclear Regulatory Commission (NRC).

7 42 U.S.C. 2011 *et seq.*

operating nuclear power plants were not members.[8]

INTO was formed in 1979 in the aftermath of the accident at Three Mile Island Nuclear Station ''to promote the highest levels of safety and reliability-to promote excellence-in the operation of nuclear electric plants. ' All commercial operators of nuclear power plants in the United States are members. INPO performs evaluations of plant practices, a form of self-regulation by peer review. The organization also conducts training and information exchange for its members. To promote effectiveness and encourage better information exchanges between member utilities, much of INPO's utility-specific work is conducted as private transactions with its members,[10] although some of its reports are provided to the NRC on a confidential basis.[11] Some INPO activities address aging-related issues, such as promoting excellence in maintenance practices, performing regular, onsite evaluations of plant facilities and practices, and analyzing operating events.

NUMARC, formed in 1987, acts as a liaison between the nuclear power industry and the NRC and other safety regulators on generic regulatory and technical issues. All U.S. nuclear utilities are members. Other nuclear industry organizations such as nuclear steam supply system vendors and architect-engineering firms also participate in NUMARC efforts. The organization has played an active role in addressing nuclear power plant aging safety issues, including major contributions in the development and implementation of NRC's maintenance and license renewal rules.

Professional societies such as the American Society of Mechanical Engineers (ASME), the Institute for Electrical and Electronics Engineers (IEEE), the American Society of Civil Engineers, and American Society of Testing and Materials have developed codes and standards for the design, maintenance, and analysis of various SSCs. Code-writing committees affiliated with these societies include individuals from utilities, vendor firms, consultants, academia, and the NRC. Several codes developed by these societies for SSC design, qualification, and maintenance have been incorporated in NRC rules.

The public and State governments also have a role in promoting the safety of existing nuclear power plants. As required by the AEA and the Administrative Procedure Act, as amended,[12] the NRC solicits public comment when developing new rules and regulations. The contribution is often extensive. For example, NRC's draft rule for nuclear power plant license renewal drew nearly 200 sets of comments, including 75 from individuals, 42 from manufacturing and engineering firms, 40 from utilities and utility organizations, 19 from public interest groups, 8 from State

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[8] Nonmembers as of December 1992 (and the number of nuclear power plants operated by each) include Commonwealth Edison (12 units); Virginia Power (4 units); Southern California Edison (2 units); Indiana and Michigan Electric (2 units); Detroit Edison (1 unit); Kansas Gas and Electric (1 unit); and Washington Public Power Supply System (1 unit). *Electric Power Research Institute 1992 Annual Report* (Palo Alto, CA: 1993), pp. 36-40; and U.S. Department of Energy, *Nuclear Reactors Built, Being Built, or Planned: 1991,* DOE/OSTI-8200-R55 (Washington, DC: July 1992).

[9] Institute of Nuclear Power Operations, "*Institutional* Plan for the Institute of Nuclear Power Operations, 1990, " p. 5.

[10] Ibid., app. B.

[11] This practice has drawn some criticism. For example, according to the U.S. General Accounting Office (GAO), on at least 12 occasions during 1989 and 1990 the NRC decided not to issue publicly available information notices after it was given access to INPO documents that were unavailable to the public. U.S. Congress, General Accounting Office, *NRC's Relationship With the Institute of Nuclear Power Operations, GAO/RCED-91-122* (Gaithersburg, MD: May 1991), p. 7. One public interest group filed a legal suit to gain access to INPO documents. However, NRC'S practice of using confidential information has been affirmed by the U.S. Court of Appeals, finding no "reason to interfere with the NRC's exercise of its own discretion in determining how it can best secure the information it needs. United States Court of Appeals, Critical Mass Energy Project, Appellant, v. Nuclear Regulatory Commission et al., 975 F.2d 871 (D.C. Cir. 1992), Aug. 21, 1992.

[12] 5 U.S.C. 551 *et seq.*

agencies, and 4 from other Federal agencies.[13] These comments led to several substantive revisions in the proposed rule.[14] Similarly, comments on the NRC's proposed maintenance rule[15] also led to changes prior to its final promulgation in July 1991.

The public may participate in NRC licensing actions associated with operating nuclear power plants that the NRC or the licensee initiates, although some observers have suggested that NRC policies have been too restrictive for public input to help address many important safety issues.[16] When a reactor licensee formally requests a modification or renewal of its NRC license, the public may request a hearing and intervene in the case, subject to certain administrative restrictions. For example, the public may request a hearing in the case of a license renewal application, but the scope of such hearings is limited to circumstances unique to the renewal term.[17] The ultimate effect of public input during NRC's deliberations over license renewal applications remains to be seen and is likely to vary by plant. Past experience with new plant licensing indicates that the role of both local and national public interest groups can be substantial.[18]

In contrast to the extensive opportunities for public participation in the development of new rules or during licensing actions initiated by the NRC or licensees, NRC regulations place strict limits on the public's ability to initiate proceedings to modify, suspend, or revoke a license. NRC regulations allow any person to file an enforcement petition with the Executive Director for Operations (EDO), a member of the NRC staff, specifying the action requested and the basis for the request. The EDO's decision in the case is subject to review of the Commission, although ''No petition for Commission review of a Director's decision will be entertained by the Commission."[19] These restrictions on petitioners' opportunities to seek Commission and judicial review have been criticized as limiting the public role in assuring plant safety. Although the requests in most public petitions have been denied, they can have notable impacts, as in the case of Yankee Nuclear Power Station (box 2-D).

Because of the technical complexity of many nuclear power issues, and because the perspectives of stakeholders can differ substantially, resolving differing opinions when new issues are raised can involve a lengthy process. For example, in developing and implementing license renewal policies, the NRC and the commercial nuclear power industry are reviewing the experience of lead plants and other related industry efforts before detailed renewal practices are finalized.

Some observers suggest that the regulatory process itself is overly cumbersome and exacerbates uncertainty.[20] According to one NRC survey

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13 U.S. Nuclear Regulatory Commission, *Analysis of Public Comments on the Proposed Rule on Nuclear Power Plant License Renewal*, NUREG-1428 (Washington DC: Oct. 5, 1991), Appendix A.

14 56 *Federal Register 64943 et seq.* (Dec. 13, 1991).

15 10 CFR 50.65.

16 Michelle Adato, The Union of Concerned Scientists, *Safety Second: The NRC and America's Nuclear Power Plants* (Indianapolis, IN: Indiana University Press, 1987).

17 This limitation is consistent with NRC's principles for the license renewal rob-that the current licensing basis provides adequate protection of the public health and safety.

18 U.S. Congress, Office of Technology Assessment, *Nuclear Power in an Age of Uncertainty*, OTA-E-216 (Washington, DC: U.S. Government Printing Office, February 1984), ch. 8.

19 10 CFR 2.206.

20 M.W. Golay, "How Prometheus Came to be Bound: Nuclear Regulation in America," *Technology Review*, June/July 1980, pp. 29-39. Although the article was written some time ago, the author contends that most of its themes remain pertinent. Personal communication January 1993.

Box 2-D-Yankee Rowe

Until its early retirement in February 1992, the Yankee Rowe nuclear powerplant, a relatively small (185 MW) PWR in Massachusetts, was the Nation's oldest operating plant. The plant began operation in 1960 and was expected to be the first to file an NRC license renewal application. During an NRC staff review of license renewal documents, questions about the ability of the pressure vessel to withstand a pressurized thermal shock (PTS) were raised.

In a petition filed with the NRC, the Union of Concerned Scientists asked for an immediate shutdown of the plant.[1] The petition emphasized several factors previously identified by NRC staff in its license renewal efforts. Yankee Rowe's case raised unique concerns related to the plant's age. For example, the pressure vessel was constructed before the susceptibility to neutron embrittlement of steel containing copper and nickel was fully understood. As a result, those elements may have been included in the vessel's weld material, although the extent of their presence was unknown. Further, due to the unique cladding of the vessel, ultrasonic testing of the vessel for cracks or flaws was not possible using conventional techniques.

Although shutdown request by the Union of Concerned Scientists was denied, the NRC initiated a review of the plant's PRA, which ultimately found that because of the uncertainties, the risk may have been greater than previously estimated.[2] The NRC revised its analysis to reflect the postulated detrimental effects of the vessel's metal cladding and made more conservative assumptions of potential cracks and the density of flaws in the vessel and welds. The NRC staff recommended shutting the plant until testing of actual plant conditions could be performed and the uncertainties resolved. This testing would involve applying specialized methods for obtaining samples of the weld materials, and for positioning ultrasonic testing equipment in the 2-inch gap between the vessel and cladding. Yankee Atomic Electric Co. concluded that the novel testing methods necessary to verify the integrity of the reactor vessel, estimated to cost $23 million, were not economically justified and voluntarily removed the plant from service and officially retired it 4 months later.[3]

1 Union of Concerned Scientists, letter to U.S. Nuclear Regulatory Commission, "Petition for Emergency Enforcement Action and Request for Public Hearing Before the Nuclear Regulatory Commission," June 4, 1991.

2 U.S. Nuclear Regulatory Commission, *In the Matter of Yankee Atomic Electric Company: Memorandum and Order, CLI-91-11,* July 31, 1991.

3 "NRC Staff, Yankee Atomic Continue Reactor Safety Debate," *The Energy Daily,* Oct. 4, 1991, p. 4.

of industry members, respondents noted that "licensees acquiesce to NRC requests even if the requests require the expenditure of significant licensee resources on matters of marginal safety significance. Further, survey respondents noted that the "NRC so dominates licensee resources through its existing and changing formal and informal requirements that licensees believe that their plants, though not unsafe, would be easier to operate, have better reliability, and may even achieve a higher degree of safety, if they were freer to manage their own resources."[21]

SAFETY PRACTICES ADDRESSING AGING

The practices necessary to manage nuclear power plant aging are elaborate, beginning with plant design and analysis and extending to a variety of maintenance and research activities. This section reviews the safety practices used to manage plant aging.

21 U.S. Nuclear Regulatory Commission, *Industry Perceptions of [he Impact of the US. Nuclear Regulatory Commission on Nuclear Power Plant Activities,* draft report, NUREG-1395 (Washington DC: March 1990), pp. xxix.

Nuclear Power Plant Design and Aging

Aging management begins with plant design. Many design criteria explicitly or implicitly address aging. The long-lived SSCs in a nuclear plant, for example, were originally designed with sufficient margins to meet minimum lifetime requirements. Nuclear power plant piping systems are designed with industry codes based on assumed service conditions, with some allowance for pipe wall thinning from erosion and corrosion. In addition, fatigue analyses used to establish design criteria for piping, pumps, and valves estimate the number of on/off cycles a power plant experiences during its life, as well as the resulting temperature variations and thermal stresses from those cycles.

To account for a variety of engineering uncertainties at the time of plant design, original SSC designs were generally based on what were then thought to be conservative assumptions of operating and material conditions.[22] Since the early plants were designed and fabricated, decades of experience and research have determined that some design assumptions were in fact not conservative, while others were. As this experience suggests, aging degradation rates for SSCs are in some cases quite different than originally anticipated.

Over the past several decades, improvements in analytical and material examination techniques have allowed the review of original plant design bases for more accurate assessments of aging degradation. More accurate predictive methods may allow for less conservatism in assessing the adequacy of SSC performance and predicting their remaining useful life. Some plants, particularly older ones, may lack the information needed for more accurate analyses. At Yankee Rowe, for example, the amount of copper in the RPV weld material was unknown, preventing any ready determinations of potential embrittlement problems. Many utilities have programs to improve the availability and retrievability of design information, including efforts to reconstitute design documents that were not adequately preserved.[23]

Technical understanding, industry practices, and NRC design requirements have become more rigorous since the 1960s. Prior to 1967, Atomic Energy Commission (AEC)[24] nuclear power plant regulations contained relatively sparse design detail. In 1971, the AEC adopted "General Design Criteria (GDC) for Nuclear Power Plants," now contained in 10 CFR Part 50, Appendix A. The GDC established minimum requirements for materials, design, fabrication, testing, inspection, and certification of all important plant safety features. The next year, a draft "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" provided more detailed guidance and requirements for implementing the GDC.[25] Additional guidance was contained in the Standard Review Plan, originally released in 1975 and revised in 1981.[26]

Codes from professional societies that are incorporated by reference in NRC regulations have also changed substantially over the past decades. For example, ASME codes for pressure vessel design, fabrication, and operating limits[27] evolved considerably from the 1960s through 1973, and in-service inspection requirements[28]

22 ASME Section XI Task Group on Reactor vessel Integrity Requirements, *White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions,* EPRI *TR-1OO251* (Palo Alto, CA: Electric Power Research Institute, January 1993), pp. 1-1 to 1-12.

23 Nuclear Management and Resources Council, *Design Basis Program Guidelines,* NUMARC *90-12 (Washington* DC: October 1990); and U.S. Nuclear Regulatory Commission, "Design Document Reconstitution" SECY-91-364, Nov. 12, 1991.

24 Regulatory authority and responsibilities were transferred to the NRC by the Energy Reorganization Act of 1974 (Public Law 93-438).

25 U.S. Nuclear Regulatory Commission Regulatory Guide 1,70 is the final version of that draft document.

26 U.S. Nuclear Regulatory Commission, *Standard Review Plan*, NUREG-0800 (Washington, DC: July 1981).

27 American Society of Mechanical Engineers, ASME Code, Section III.

28 American Society of Mechanical Engineers, ASME Code, Section XI.

were introduced in 1970.[29] Similarly, IEEE standards for electrical equipment issued in 1971 were substantially revised in 1974.[30]

Some observers have suggested that the safety of older plants is inadequate, because they were not designed with the same detailed guidance as newer plants and therefore often do not meet the current design standards.[31] However, the commercial nuclear power industry and the NRC note that the NRC judges safety for older plants on an ad hoc and plant-specific basis, rather than a standardized basis, and the NRC finds that adequate safety currently exists. To review and ensure the safety of older plants, the NRC created the "Systematic Evaluation Program" (SEP) in 1977. According to the NRC, the SEP review of approximately 90 topics necessitated some specific procedural or hardware modifications (''backfits' '), and additional analyses for the older plants provided "reasonable assurance that they can be operated without undue risk to the public health and safety, which is the same standard for new plants.[32]

Maintenance Practices Addressing Aging

Effective maintenance programs are crucial to manage aging degradation. Maintenance involves a variety of methods to predict or detect aging degradation and other causes of SSC failure, and to repair or replace any affected SSCs. Both NRC rules and industry codes contain maintenance requirements. For example, the ASME Boiler and Pressure Vessel Code Section XI specifies inservice inspection methods, which are incorporated in NRC rules.[33] Before 1991, there were no specific NRC maintenance requirements for many SSCs important to safety. To ''ensure the continuing effectiveness of maintenance for the lifetime of nuclear power plants, particularly as plants age, " the NRC adopted a maintenance rule in 1991 to become effective in 1996.[34] The rule directs Licensees to establish performance goals for SSCs important to safety and to monitor the condition or performance of those SSCs, or otherwise control degradation through preventive maintenance. The requirements are relatively flexible and do not specify performance criteria (e.g., the frequency of testing or surveillance), and the rule does not require a detailed regulatory approval of the criteria licensees establish.

The maintenance rule was promulgated after several years of increasing NRC and industry attention to maintenance.[35] While the NRC was evaluating the need for a maintenance rule, INPO developed guidelines for effective maintenance to guide utility practices.[36] As a result, the industry argued that the NRC rule was unnecessary and duplicated current practices established by INPO. In promulgating its rule, the NRC noted that its recent inspections of maintenance activities found that existing programs were adequate and improving, but there were some areas of weaknesses, and NRC found that no licensee had formally committed to implement the INPO

29 ASME Section XI Task Group on Reactor vessel Integrity Requirements, *White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions,* EPRI TR-100251 (Palo Alto, CA: Electric Power Research Institute, January 1993), pp. 1-1 to 1-12.

30 Institute of Electrical and Electronics Engineers Standard, "Criteria for Protection Systems for Nuclear Power Generating Stations, " (IEEE-279), incorporated by reference in 10 CFR Part 50.55a(h).

31 See, e.g., Diane Curran, Counsel for Union of Concerned Scientists, *Hearings Before the Subcommittee on Energy and the Environment of the Committee on Interior and Insular Affairs*, House of Representatives, Nov. 5, 1991, pp. 93-95.

32 U.S. Nuclear Regulatory Commission, *Foundation for the Adequacy of the Licensing Bases,* NUREG-1412 (Washington, DC: December 1991), p. 1.5.

33 10 CFR 50.55a.

34 10 CFR 50.65.

35 U.S. Congress, General Accounting Office, NRC'S *Efforts to Ensure Effective Plant Maintenance Are Incomplete* GAO/RCED-91-36 (Gaithersburg, MD: December 1990).

36 Institute of Nuclear Power Operations, ''Maintenance Programs in the Nuclear Power Industry,' INPO 90-008, March 1990.

standards prior to the rule's proposal.[37] NUMARC later submitted the INPO guidelines to the NRC as an industry standard suitable for compliance with the maintenance rule. The group also coordinated a validation and verification effort of the maintenance approach at several nuclear plants, and the NRC found them to describe adequately the attributes necessary to comply with the maintenance rule.[38]

There have been significant advances in nuclear plant maintenance technologies in the last two decades in all areas, including surveillance, testing, and inspection of important SSCs subject to degradation; methods to plan repair, replacement, and other maintenance activities; and actual SSC repair and replacement methods. All are important to ensure that aging degradation does not unduly reduce plant safety margins and performance. There is a wide variety of specific inspection, surveillance, testing, and monitoring techniques used for the many different plant SSCs. Examples of improved maintenance techniques for two major long-lived SSCs are given in boxes 2-B (RPV embrittlement) and 2-C (steam generator tube corrosion and cracking).

Effective maintenance requires the careful planning and design of maintenance programs. Two areas of improved planning approaches are: 1) predictive and reliability-centered maintenance (RCM)[39]; and 2) risk-focused maintenance (RFM). RCM involves the use of prediction and inspection techniques to repair or replace degraded critical equipment prior to its failure.[40] Absent a reliability-based approach, much maintenance work focuses on either repairing failed equipment as it occurs or repairing or replacing equipment long before it wears out. In addition to placing heavy reliance on the defense-indepth approach designed into nuclear plants (e.g., redundancy of important safety items), reactive maintenance in the extreme results in more plant shutdowns and less coordination of maintenance with fuel cycle outages. At the other extreme, premature replacement of properly functioning SSCs represents an unnecessary cost and increase the potential for maintenance errors. RCM involves inspection and repair before SSCs wear out but avoids excessive repair work through monitoring and predictive techniques. The RCM concept continues to evolve, for example, in selecting an appropriate level of detail (e.g., to examine systems or individual components) .41 RCM efforts, involving either pilot programs or significant investments, are under way at about half of the nuclear plants in the United States.[42] NRC's regulatory guide for the maintenance rule encourages utilities to consider reliability-based methods of predictive maintenance.[43]

RFM uses probabilistic risk assessment (PRA) methods to determine which SSCs subject to degradation are most important to safety and performance and thus which should receive the greatest maintenance attention.[44] For example, rather than perform an equal number of tests or inspections on all of the many valves in a nuclear power plant, those most important to reducing or

37 56 FR 132, July 10, 1991, p. 31321.

38 Nuclear Management and Resources Council, *Industry Guidelines for Monitoring Effectiveness of Maintenance at Nuclear Power Plants,* NUMARC *93-01 (Washington DC: May 1993); 56 Federal Register 31312* (July 10, 1991); and U.S. Nuclear Regulatory Commission, Regulatory Guide 1.160, June 1993.

39 NUS Corporation, Predictive Maintenance *Primer,* EPRI NP-7205 (Palo Alto, CA: Electric Power Research Institute, April 1991).

40 For equipment not critical to safety, the prescribed maintenance approach may well be one of running until failure.

41 "NUMARC Wants N. Utilities Moving Early on Maintenance Rule Work," *Nucleonics Week,* vol. *33, No.* 42, Oct. 15, 1992, pp. 1, 13.

42 D.H. Worledge, "Nuclear Industry Embraces Reliability-Centered Maintenance,' *Power Engineering*, July 1993, pp. 25-28.

43 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.160, June 1993.

44 E.V. Lofgren et al., A Process *for Risk-Focused Maintenance,* NUREG/CR-5695 (Washington, DC: U.S. Nuclear Regulatory Commission, March 1991).

mitigating accident risks are inspected more frequently. RFM is also applied to EDG testing; during any cold start for an engine such as a diesel generator, the thermal stresses and mechanical wear from the initial lack of lubrication contributes to substantial degradation and the potential for premature failure. One RFM application has allowed plant operators to reduce the frequency of cold start EDG testing, while increasing the testing of other emergency generator components and support equipment, such as the starter systems. The result: longer and more reliable lives for the EDGs and a higher expected availability when they are actually needed.

Degradation detection methods for many SSCs typically have imperfect accuracy,[45] a factor to consider when designing maintenance practices. Improved testing and inspection techniques continue to be developed, allowing more accurate and earlier detection of flaws and other material characteristics, and improving the likelihood of preventing the failure of important SSCs. New nondestructive examination (NDE) methods—including ultrasonic, eddy-current, and radiographic inspections of pressure vessels, steam generators, piping, containment and other SSCs--allow more accurate SSC evaluations than previously possible.[46] For example, new NDE methods based on magneto-optic imaging allow examination of containment welds for cracking, even when these welds are beneath paint coatings.[47]

In addition, new methods are under development to examine some important SSCs that currently preclude testing or inspection due to basic physical limitations (e.g., limited access or space). New robotic technologies and other specialized inspection machines allow better access to confined or high radiation areas.[48] Robotics applications include underwater visual inspections using submersible vehicles with cameras, internal inspection of piping using power crawlers, and cleaning RPV internals and steam generators. After detecting cracks in RPV head penetrations at its Bugey-3 nuclear power plant, for example, Electricity de France (EDF) decided to inspect these penetrations at all 59 of its pressurized water reactors (PWRs). To reduce the substantial occupational exposures resulting from the detailed inspections, EDF worked with equipment vendors to develop a specialized robotic inspection device to reduce exposures substantially.[49] The use of robotics in maintenance activities is increasing, but improvements in precision, dexterity, and mobility could increase their usefulness further.

Unanticipated degradation rates have inspired new repair and replacement methods for some major SSCs. In some cases, such as with some PWR steam generators and boiling water reactor (BWR) recirculation piping, these methods have become widespread. However, replacing or repairing some SSCs may not be economically or technically practical. Even where replacement or repair is infeasible, life-limiting challenges may be addressed through revised O&M practices; such changes may reduce stresses on a vulnerable SSC or may involve more regular monitoring to detect incipient failure.

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45 See, for example, Pacific *Northwest Laboratory, Ultrasonic Inspection Reliability for Intergranular Stress Corrosion Cracks,* NUREG/CR-4908 *(Washington DC:* U.S. Nuclear Regulatory Commission, July 1990).

46 J.A. Jones Applied Research *Co., Nondestructive Evaluation Sourcebook,* EPRI NP-7466-M (Palo Alto, CA: Electric Power Research Institute, September 1991).

47 Physical Research, Inc., Two *New* NDT *Techniques for Inspection of Containment Welds Beneath Coating, NUREG/CR-5551* (Washington DC: U.S. Nuclear Regulatory Commission, June 1991).

48 Utility/Manufacturers Robot Users Group, *Survey of Utility Robotic Applications (1990),* EPRI NP-7456 (Palo Alto, CA: Electric power Research Institute, August 1991).

4$1 ''Nuclear Industry Deflects Greenpeace on Cracking Issue, " *Nucleonics Week,* vol. *34, No. 13,* Apr. 1, 1993, pp. 1,9-12. The U.S. nuclear power industry and the NRC expect to begin detailed inspections of PWR RPV head penetrations in 1994 when specialized machines become available.

Maintenance technologies continue to evolve, and greater experience and implementation hold the promise of safer, more reliable, and less costly operations. To transfer the results of maintenance R&D, EPRI has established a Nuclear Maintenance Applications Center in North Carolina.[50] The Center provides a forum to impart EPRI research findings and assists with training and information exchange for nuclear utilities.

Aging Research

Both the commercial nuclear power industry and the NRC view continued aging research and analysis of operating experience as important to help assure adequate safety. Both the industry and the NRC perform research on a broad range of aging topics, including basic materials science, studies of specific components and degradation mechanisms, new maintenance practices, and analytical techniques.

Since its inception in 1973, EPRI has devoted about 15 percent of its Nuclear Power Division budget to understanding, detecting, and mitigating degradation processes for nuclear power plant components.[51] The 1992 EPRI R&D plan included over $130 million in nuclear power activities.[52] Similarly, the AEC and its successory, the NRC, have conducted research on materials aging since 1960. About 25 percent of the current $100 million annual NRC research budget is dedicated to aging research.[53] Most NRC aging research is performed through Department of Energy (DOE) national laboratories. Aging research is also conducted by some international organizations and other nations with nuclear power plants.[54]

The goals of safety-related aging research are varied and include the following:

- understanding SSC aging effects that could impair plant safety if unmitigated;
- developing inspection, surveillance, monitoring, and prediction methods to ensure timely detection of aging degradation;
- evaluating the effectiveness of operating and maintenance practices to mitigate aging effects; and
- providing the technical bases for license renewal.[55]

Absent actual, long-term operating experience for long-lived SSCs, scientific understanding of aging issues involves engineering analyses and research, often using simulation techniques to accelerate aging on test materials.[56] Retired plants may also yield lessons about aging by providing naturally aged SSCs to study. For example, the NRC, the DOE and the commercial nuclear power industry are coordinating efforts to examine materials from the retired Yankee Rowe plant, which operated for 30 years, to aid in aging research.[57] However, the diversity among plants and their SSCs prevents simple generalizations about the ultimate effects and management of aging. In contrast, for shorter lived SSCs, engi-

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50 See, for example, Electric Power Research Institute, *EPRI Research Publications, Products, and Expertise in Maintenance,* EPRI NP-7014 (Palo Alto, CA: May 1991).

51 John Carey, Electric Power Research Institute, personal communication, January 1993.

52 See, for example, Electric Power Research Institute, *Research and Development Plan 1993,* (Palo Alto, CA: 1993).

53 U.S. Nuclear Regulatory Commission, *Budget Estimates Fiscal Years 1994-1995,* NUREG 1100, vol. 9 (Washington, DC: April 1993), pp. 48, 51.

54 See International Atomic Energy Agency, *Safety Aspects of the Aging and Maintenance of Nuclear Power Plants,* (Vienna, Austria: 1988); and International Atomic Energy Agency, *Safety Aspects of Nuclear Power Plant Ageing,* IAEA-TECDOC-540 (Vienna, Austria: 1990).

55 U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) program Plan,* NUREG-1144 Rev. 2 (Washington, DC: U.S. Nuclear Regulatory Commission, June 1991), p. 1.4.

56 University of Connecticut, *Natural Versus Artificial Aging of Nuclear Power Plant Components,* EPRI TR-100245 (Palo Alto, CA: Electric Power Research Institute, January 1992).

57 58 *Federal Register 8998-8999* (Feb. 18, 1993).

neering analyses and aging research are supported better by actual operating experience.

According to the NRC, "there are significant uncertainties about aging degradation processes and about whether time-related degradation can be detected and managed before safety is impaired." [58] However, no incurable safety problems have yet been identified by NRC aging research studies. Rather, NRC research has improved the understanding of aging issues and the adequacy of maintenance efforts. These research findings are transferred to NRC regulatory activities, including plant inspections and revisions of technical specifications.[59] Figure 2-3, which shows the results of research on BWR recirculation piping, provides one example of how information gained from aging research has influenced regulatory and operating practices. As of 1991, the NRC anticipated the completion of its Nuclear Plant Aging Research (NPAR) program as currently formulated by 1997 ('box 2-E), although that schedule is not firm (tables 2-1 and 2-2).[60] Even with the completion of the NPAR program, research will be needed to examine new maintenance methods and to address any new issues identified through operating experience and past research.

The results of generic SSC aging evaluations relevant to license renewal are documented in 10 industry reports produced with industry and DOE funds. The reports were produced by EPRI and DOE's Sandia National Laboratory for NUMARC, and NUMARC submitted them to the NRC for an evaluation of their applicability for utilities submitting renewal applications. These reports are intended to examine all plausible aging degradation mechanisms, and identify combinations of components and degradation mechanisms for which existing programs do not effectively manage the degradation. Consistent with the results of NRC's research, this effort identified no incurable safety challenges, and found that most component degradation mechanisms are effectively managed by current plant programs. However, plant-specific challenges may exist, and several areas for further examination were identified. As with much of NRC's aging research, these documents are generic rather than plant-specific.

External Review of Nuclear Power Plant Activities

Regular external review of nuclear utility power plant and corporate activities in the form of safety inspections and evaluations is fundamental to ensure safety for plants of all ages.[61] Outside inspections and evaluations of licensee performance are conducted by both the NRC and INPO. Some external review activities are closely related to concerns about plant aging. For example, reviews of utility maintenance practices can help ensure that those activities are performed adequately and will effectively identify degradation related to aging or other causes.

INPO evaluations of operating plants and corporate organizations involve in-depth team reviews conducted at an average interval of about 16 months.[62] The INPO evaluation reports are provided to the utility and are available to the NRC resident inspector but are not public documents. Subsequent INPO evaluations assess the

58 U.S. Nuclear Regulatory Commission, *Annual Report 1991,* NUREG-1 145, vol. 8 (Washington, DC: July 1992), P. 161.

59 W. Gunther and J. Taylor, Brookhaven National Laboratory, *Results from the Nuclear Plant Aging Research program: Their Use in Inspection Activities,* NUREG/CR-5507 (Washington, DC: U.S. Nuclear Regulatory Commission September 1990); and U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research Program Plan,* NUREG-1 144, Rev. 2 (Washington, DC: June 1991), pp. 6.23-6.33,

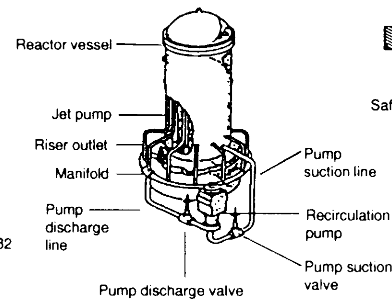
60 Lawrence Shao, Director, Engineering Division, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, personal communication, February 1993.

61 U.S. Nuclear Regulatory Commission, *Annual Report 1991*, NUREG-1 145, vol. 8 (Washington DC: July 1992), pp. 19-25.

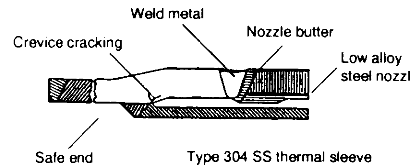
62 Institute of Nuclear Power Operations, "Institutional Plan for the Institute of Nuclear' Power Operations," Appendix A.

Understanding and managing aging of BWR recirculation piping

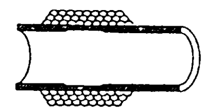
Materials	Piping	Types 304, 316L, and 316NG SS	
	Fittings, pumps and valves	Statistically cast SS - Gr. CF8 and CF8M	
	Safe ends	Types 304, 316, 316NG SS and Alloy 600	
	Pipe-to-pipe and pipe-to-elbow welds	Type 308L SS	
	Safe end-to-nozzle weld		
	Weld material	308L/308 SS or	Alloy I-182/I-82
	Nozzle butter	308L SS	Alloy I-182
	Safe end butter	None	Alloy I-182
Stressors and environment	Operational transients, residual tensile stresses, applied stresses, oxygenated coolant, impurities in coolant, and contaminants on the outside surface.		



BWR recirculation piping system



Crack at recirculating water inlet nozzle thermal sleeve junction



Typical weld overlay reinforcement

UNDERSTANDING AGING
(materials, stressors, and environmental interaction)

SITES	AGING CONCERNS
vveld sensitized heat-affected zone. Furnace sensitized safe end, bimetallic and trimetallic welds (Types 304, 304L, 316, 316L SS).	GSCC · inside suface · outside surface, if high residual tensile stresses (may be introduced, for example, by stress improvement) and contaminants are present.
Sites with crevices and small amount of cold work	GSCC
Sites with highly stressed regions and large amount of cold work (Type 316NG SS).	Transgranular stress corrosion cracking (TGSCC).
Sites subject to cyclic stresses.	Fatigue
Duplex (austenitic-ferritic) stainless steels: Fittings, pump and valve castings.	Thermal embrittlement of high delta ferrite (greater than 10 percent ferrite) regions, IGSCC of low delta ferrite (less than 10 percent ferrite) regions.

MANAGING AGING

SERVICE INSPECTION, SURVEILLANCE & MONITORING

NRC requirements

Ultrasonic inspection of weldments as per NRC Generic Letter 88-01.

Reduced inspection period and increased sample size for piping made of non-IGSCC resistant materials (e.g., Type 304 SS):

- for weldments without cracks, depends on when stress improvement implemented;
- for weldments with cracks, depends on whether cracks have been reinforced by weld overlay or mitigated by stress improvement;
- inspection procedure for weld overlay repair to detect cracks that were 75 percent of the original wall thickness;
- inspection period can be as short as one refueling cycle, and sample size can be as large as all welds.

Recommendations

Develop improved methods to inspect weld overlay repairs.

For leak detection:
- install moisture sensitive tape on susceptible region of piping
- use acoustic emission monitoring method.

Conduct online monitoring of coolant chemistry.

For effective hydrogen water chemistry control:
- maintain electrochemical corrosion potential on standard hydrogen electrode scale to less than or equal to -230 mV
- maintain coolant conductivity less than or equal to μS/cm.

MITIGATION

Employ weld overlay, mechanical stress improvement, induction heating stress improvement, and clamping device for repairing welded piping.

Use Type 316NG SS corrosion-resistant cladding, solution heat treatment, and heat sink welding or replacement piping.

Evaluate the use of 347 NG SS as piping material.

Assess the effects of hydrogen water chemistry on overall plant operation:
- radiation fields;
- fuel performance; and
- acceptable length and frequency of short outages of hydrogen water chemistry.

Figure 2-3-Nuclear Plant Aging Research Program Summary of Boiling Water Reactor Recirculation Piping Aging Issues

SOURCE: U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) Program Plan,* NUREG-1 144, Rev. 2, June 1991, p. 6.32.

Box 2-E—The Nuclear Regulatory Commission's Nuclear Plant Aging Research Program

Under the U.S. Nuclear Regulatory Commission's (NRC's) Nuclear Plant Aging Research (NPAR) Program, aging assessments have been or are being performed on over 40 categories of systems, structures, and components (SSCs) considered significant to safety, many of which are relatively short-lived.[1] These SSCs were selected based on their significance to plant safety, operating experience, expert opinion, and susceptibility to aging degradation, not necessarily whether they are short- or long-lived.

A one- or two-phase examination is performed for each SSC. Phase I involves a paper examination, including review of the design, materials, and operating stresses and a survey of operating experiences and historical failures for the selected SSC. Also, the existing SSC inspection and monitoring methods are examined to determine their effectiveness in detecting aging degradation before failure occurs. Often, the adequacy of artificial or accelerated aging techniques used to qualify the SSC for its design lifetime are compared to available data from their naturally aged counterparts. The result of a Phase I evaluation is an interim assessment of probable failure modes.

Phase-II NPAR assessments, which the NRC may deem unnecessary depending on Phase I results, may involve laboratory tests of naturally or artificially aged equipment; aging assessments by experts; recommendations for inspection or monitoring techniques; and in-situ examinations. As shown in the tables, analyses have been performed for many SSCs, but several have yet to be initiated.

Because of substantial variations in hardware and procedures at U.S. operating nuclear plants, the NRC examinations are not intended as in-depth engineering evaluations of all significant SSCs. That responsibility ultimately belongs to the operators of each nuclear plant. This is particularly the case with major components and structures such as pressure vessels, emergency diesel generators (EDGs), or primary containment, for which laboratory examinations are infeasible.

For example, nuclear power plant EDGs are large and complex, with about 25 models supplied by nine vendors in current use. Because naturally aged EDGs on which to perform indepth laboratory examinations are not available, the NPAR program approach is to use expert opinion drawn from national laboratories, consultants, manufacturers, and utilities to examine historical failures and to identify the components most vulnerable to aging and identify mitigation measures.

[1] Structural and materials aging research are conducted under separate programs at the NRC.

[2] K.R. Hoopingarner and F.R. Zaloudek, Pacific Northwest Laboratory, *Aging Mitigation and Improved Programs for Nuclear Service Diesel Genertors* NUREG/CR-5057 (Washington, DC: U.S. Nuclear Regulatory Commission, March 1989).

Table 2-l-Systems and Components in the Nuclear Plant Aging Research Program and Their Completion Schedule

Topic	Laboratory	Schedule
Components		
Motor-operated valves	ORNL	Complete in fiscal year 1991
Check valves	ORNL	Complete in fiscal year 1991
Solenoid valves	ORNL	Complete in fiscal year 1991
Air-operated valves	ORNL	initiate Phase 1 in fiscal year 1991
Auxiliary feedwater pumps	ORNL	Complete in fiscal year 1991
Small electric motors	ORNL	Completed in fiscal year 1988
Large electric motors	BNL	Initiate phase 1 in fiscal year 1992
Chargers/inverters	BNL	Completed in fiscal year 1990
Batteries	INEL	Completed in fiscal year 1990
Power-operated relief valves	ORNL	Completed In fiscal year 1989
Snubbers	PNL	Complete phase 2 in fiscal year 1991
Circuit breakers/relays	BNL, Wyle	Complete phase 2 in fiscal year 1991
Electrical penetrations	SNL	Complete phase 1 in fiscal year 1991
Connectors, terminal blocks	SNL	Initiate phase 1 in fiscal year 1991
Chillers	PNL	Initiate phase 1 in fiscal year 1991
Cables	SNL	Complete phase 2 in fiscal year 1991
Diesel generators	PNL	Phase 2 completed in fiscal year 1989
Transformers	INEL	Complete phase 1 in fiscal year 1991
Heat exchangers	ORNL	Complete phase 1 in fiscal year 1991
Compressors	ORNL	Phase 1 completed in fiscal year 1990
Bistables/switches	BNL	Initiate phase 1 in fiscal year 1991
Main steam isolation valves	ORNL	Initiate phase 1 in fiscal year 1991
Accumulators		No initiative
Surge arrestors		No initiative
Isolation condensers (BWR)		No initiative
Purge and vent valves		No initiative
Safety relief valves		No initiative
Service water and component cooling water pumps		No initiative
Systems		
High-pressure emergency core cooling system	INEL	Complete phase 1 in fiscal year 1991
RHR/Low-pressure emergency core cooling system	BNL	Complete phase 2 in fiscal year 1991
Service water	PNL	Phase 2 completed in fiscal year 1990
Component cooling water	BNL	Complete phase 2 in fiscal year 1992
Reactor protection	INEL	Complete phase 2 in fiscal year 1991
Class 1 E electric distribution	INEL	Complete phase 2 in fiscal year 1991
Auxiliary feed water	ORNL	initiate phase 1 in fiscal year 1991
Control rod drive, PWR (W)	BNL	Phase 1 completed in fiscal year 1990
Control rod drive, PWR (B&W, CE)	BNL	Complete phase 1 in fiscal year 1991
Control rod drive, BWR	ORNL	Complete phase 1 in fiscal year 1991
Motor control centers	BNL	Completed in fiscal year 1989
instrument air	BNL	Complete phase 2 in fiscal year 1992
Containment cooling	BNL	Complete phase 1 in fiscal year 1991
Engineered safety features	PNL	Initiate phase 1 in fiscal year 1991
instrument and control	ORNL	Complete phase 1 in fiscal year 1992
Automatic depressurization (BWR)	PNL	Complete pre-phase 1 in fiscal year 1991
Standby liquid control (BWR)	PNL	Complete phase 1 in fiscal year 1991
Core internals	ORNL	initiate phase 1 in fiscal year 1991
Turbine main generator and controls	ORNL	initiate phase 1 in 1991
Containment isolation		No initiative
Recirculation pump trip actuation instrumentation (BWR)		No initiative
Reactor core isolation cooling		No initiative

SOURCE: U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) Program P/an,* NUREG-1144, Rev. 2 (Washington, DC: June 1991).

effectiveness of utility actions to address previously identified items.

The NRC inspection program is intended to evaluate plant compliance with the current licensing basis (CLB) (box 2-F), to determine reactor safety, and to identify conditions that may warrant corrective actions. The inspection staff also collects information used in the NRC Systematic Assessment of Licensee Performance (SALP) evaluations (box 2-G). Each operating plant has at least one full-time, onsite NRC resident inspector. The resident inspectors directly observe and verify licensee activities in the control room, in maintenance and surveillance testing, and in the configuration of equipment important to safety, and they conduct frequent general plant tours, In addition to the regular duties of resident inspectors, inspectors from the five NRC regional offices and the NRC headquarters periodically perform a variety of more detailed technical inspections.

NRC team inspections are conducted by technical specialists drawn from both the NRC and its contractor organizations (e.g., the national laboratories). These specialists spend several weeks at a plant investigating a specific topic, such as maintenance, emergency operations, or the testing of motor-operated valves. Maintenance Team Inspections in which all maintenance-related plant activities were observed in detail were conducted at all plants in the late 1980s and early 1990s. These inspections found adequate programs and implementation at all sites. These favorable findings partially explain why the NRC promulgated a relatively flexible maintenance rule in 1991.[63]

Table 2-2-Completed Nuclear Plant Aging Research Life Assessments for Major Components

Emergency diesel generators
Pressurized Water Reactor (BWR) and Boiling Water Reactor (BWR) pressure vessels
BWR Mark I containments
PWR and BWR pressure vessel Internals
PWR cooling system piping and nozzles
PWR steam generator tubes
Pressurizer, surge and spray lines
BWR recirculation piping
LWR coolant pumps

SOURCE: U.S. Nuclear Regulatory Commission, *Nuclear P/ant Aging Research (NPAR) Program P/an,* NUREG-1 144, Rev. 2. (Washington, DC: June 1991).

License Expiration and Renewal for Aging Management

The AEA specifies that commercial nuclear plant operating licenses may not exceed 40 years but may be renewed upon expiration.[64] The fixed term was established in the AEA for financial and other nontechnical reasons, although once chosen, it became an assumption in specifying certain plant design features (e.g., the number of thermal cycles occurring, and thus the requirements for addressing fatigue).

NRC license renewal requirements center on the management of aging degradation. As a result of its license renewal work, the NRC staff identified fatigue and environmental qualification of electrical equipment (EQ) as possible generic safety issues to be examined for all plants during their current license terms.[65] The importance of fatigue and EQ to aging is well known to both the commercial nuclear power industry and the NRC, and considerable attention has been directed to these issues (box 2-H). Rather than identifying new aging issues, examining these

63 56 *Federal Register* 31321 (July 10, 1991).

64 License terms were initially set based on the start of plant construction rather than the start of operation. However, the NRC h&s established a relatively simple administrative procedure to recover the construction period and thereby extend the expiration date of the initial operating licenses without renewal. Memorandum from W.J. Dircks, Executive Director for Operations to the Commissioners, U.S. Nuclear Regulatory Commission, Aug. 16, 1982. To date, over fifty such extensions have been granted. 58 *Federal Register 7899.* Feb. 10, 1993.

65 U.S. Nuclear Regulatory Co remission, Implementation of 10 CFR Part 54, ‘Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” SECY-93-049, Mar. 1, 1993.

Box 2-F-Current Licensing Bases

A plant's current licensing basis (CLB) includes all NRC requirements, whether made during initial licensing or as modified over time.[1] This **large body of** requirements is contained in a **variety of documents,** including:

- a plant's operating license application or Safety Analysis Report;
- plant-specific compliance with NRC regulations noted in 10 CFR Part 50, as well as other parts of Title 10 of the Code of Federal Regulations;
- NRC orders, license conditions, exemptions, and technical specifications; and
- all written commitments made by the licensee in docketed responses to NRC bulletins **and generic** letters.[2]

NRC regulations and industry practices draw on the codes and standards of many organizations such as the American Society of Mechanical Engineers, the Institute for Electrical and electronics Engineers, the American Society of Civil Engineers, and American Society of Testing and Materials.

The CLB for each plant is unique. Differences result from variations in plant siting (e.g., a plant located near an active fault requires special seismic protection features); plant design (e.g., whether a boiling or pressurized water reactor, the number of steam generators); different regulations and regulatory interpretations in effect at the time of licensing; and plant operating experience (e.g., special problems leading to additional commitments to the NRC). Many NRC requirements, such as the maintenance rule, explicitly address aging safety issues.

1 For additional discussion of the NRC's views of current licensing bases, see U.S. Nuclear Regulatory Commission, *Foundation for the Adequacy of the Licensing Bases*, NUREG-1412 (Washington, DC: December 1991).

2 In its effort to provide the commercial nuclear power Industry Information on operating experience, each year the NRC issues about 5 generic bulletins, about 20 generic letters, and about 100 information notices. Science Applications international Corporation, *Generic Communications Index*, NUREG/CR-4690 (Washington, DC: U.S. Nuclear Regulatory Commission, May 1991). Although the informal guidance does not carry the same legal authority as regulations, licensees are often motivated to address the issues raised Their docketed responses to the generic communications then become part of the plant's formal requirements.

topics as generic safety issues provides a method for identifying and prioritizing issues based on potential safety significance and implementation costs .66

The NRC license renewal rule is founded on two key principles:

1. With the exception of age-related degradation unique to license renewal (ARDUTLR), and possibly some few other issues related to safety only during extended operation, the existing regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety; and
2. each plant's CLB must be maintained during the renewal period, in part through a program of aging degradation management for SSCs that are important to license renewal.[67]

If approved, the renewed License would supersede the existing license, with the requested extension period increased to reflect the time remaining under the current license.

In any event, the duration of the renewal license would be limited to 40 years, including an extension of no more than 20 years. The NRC has estimated that the effort required by a utility to submit a license renewal application would re-

66 U.S. Nuclear Regulatory Commission, *A Prioritization of Generic Safety Issues*, NUREG-0933, semi-mud report series.

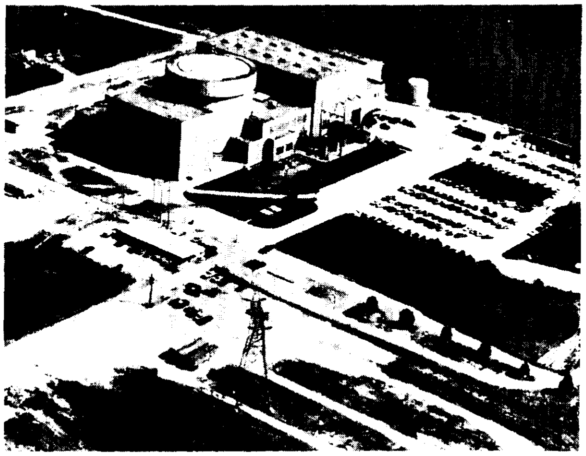
67 56 *Federal Register* 64943 *et seq.* (Dec. 13, 1991).

quire approximately 200 person-years of utility effort (supplemented by unquantified consultant support) and span 3 to 5 calendar years at a cost of about $30 million.[68]

Under the license renewal rule, an applicant must perform an integrated plant assessment (IPA), analyzing all mechanisms that result in age degradation, even for short-lived SSCs that are routinely replaced. For degradation identified as ARDUTLR, the utility must demonstrate a program to monitor or control that degradation. This plant-specific assessment is intended to guide the licensee through a structured process in order to demonstrate that aging degradation of plant SSCs has been identified, evaluated, and addressed, and to ensure that the licensing basis will be maintained throughout the renewed license term.

As discussed in detail in chapter one, there are some practical problems with implementing the rule and its accompanying statement of considerations (SOC). These involve such issues as the level of detail required in the IPA, problems with key definitions (e.g., ARDUTLR as defined has little practical meaning), and consistency with other aging management requirements (e.g., the maintenance rule). As discussed in chapter one, the NRC is considering revising the rule or specifying a simplified implementation process.[69]

No plant has yet submitted a license renewal application. Owners of the Yankee Rowe and the Monticello plants originally planned to submit license renewal applications in 1991 as part of a jointly funded, multiyear DOE/industry lead-plant program. However, poor economics, including the costs of answering questions about the safety of their RPV, prompted Yankee Rowe's owners to opt for early retirement in late 1991. In late 1992, Monticello's owner indefinitely de-



U.S. NUCLEAR REGULATORY COMMISSION

The operating license for the Fort Calhoun Station expires in 2008. Recapture of the construction period could allow 5 years additional operation before license renewal would be required.

ferred its renewal application, citing concern about the interpretation of NRC's rule, noting that the number of systems to be reviewed had grown from the original 74 to 104 with “no indication of where it might go from there. ’[70] Also noted were concern over operational cost increases and about DOE's ability to accept spent fuel. Finally, in late 1992, the Babcock and Wilcox Owners' Group announced its intentions to pursue a joint effort in developing a license renewal application. Other owners' groups are pursuing similar efforts.

License renewal has implications for other NRC safety requirements for specific plants. One example is application of the backfitting rule.[71] Although a plant's CLB is supposed to be adequate for protecting the public health and safety, the backfitting rule allows additional requirements when certain conditions are met. Specifically, the rule allows such additional requirements if a backfit analysis shows that there

68 U.S. Nuclear *Regulatory* Commission, *Regulatory Analysis for Final Rule on Nuclear Power Plant License Renewal,* NUREG-1362 (Washington, DC: October 1991), table 4.6.

69 U.S. Nuclear Regulatory Commission, *SKY-93-1 13,* Apr. 30, 1993; and U.S. Nuclear Regulatory Commission, SECY-93-049, Mar. 1, 1993.

70 Jim Howard, Chief Executive Officer of Northern States power cited in *Nucleonics Week*, vol 33, No. 46, Nov. 12, 1992, *pp.* 12,13.

71 10 CFR 50.109.

Box 2-G--Systematic Assessment of Licensee Performance and Other Performance Indicators

The Systematic Assessment of Licensee Performance (SALP) program is an integrated effort to assess how well a given licensee directs and provides the resources necessary to provide the requisite assurance of safety. The purpose of these NRC assessments is to direct better both the NRC and licensee attention and resources at a facility to those safety issues requiring the most attention. Some in the nuclear industry, however, have suggested that the SALP process is subjective and not factually supported.[1]

The SALP assessment includes reviews of licensee event reports (LERs), inspection reports, enforcement history, and licensing issues. These ratings are a subjective summary of the performance of the licensee in each functional area New data are not necessarily generated in the conduct of a SALP assessment. The SALP assessment rates performance in selected functional areas: plant operations, radiological controls, maintenance and surveillance, emergency preparedness, security, engineering and technical support, and safety assessment and quality verification. SALP rating categories are the following:

1: This rating designates a superior level of performance where reduced NRC attention maybe appropriate.

2: This rating designates a good level of performance where NRC attention should be maintained at normal levels.

3: This rating designates an acceptable level of performance where the NRC will consider increased levels of inspection.

N: insufficient information exists to support an assessment of licensee performance.

NOTE: There is no failing grade, but plants not meeting acceptable levels (i.e., inadequate performance to receive a category 3 rating) are issued a "show cause" order resulting In thdr shutdown.

Since 1986 the NRC has also provided quantitative indicators of nuclear power plant safety performance. The program currently provides seven performance indicators, including the average number of SCRAMS and the equipment forced outage rate (see figures 2-4 and 2-5). These data are published and provided to NRC senior managers on a quarterly basis, and each utility receives the reports for its plants. in contrast with the NRC SALP program, which provides subjective evacuations of licensee performance, the performance indicators measure well-defined, discrete events. However, the relationship between these indicators and expected public health and safety impacts, while giving a sense of safety performance, is not definitive.

The Institute of Nuclear Power Operations (INPO) has also developed quantitative indicators of nuclear performance. The INPO program includes such factors as plant capability factor, rate of unplanned automatic scrams, collective radiation exposure, and industrial accident rates. in addition to publishing the indicators for industry-wide performance, INPO has set goals for improving future performance that are intended to be challenging but achievable.[2]

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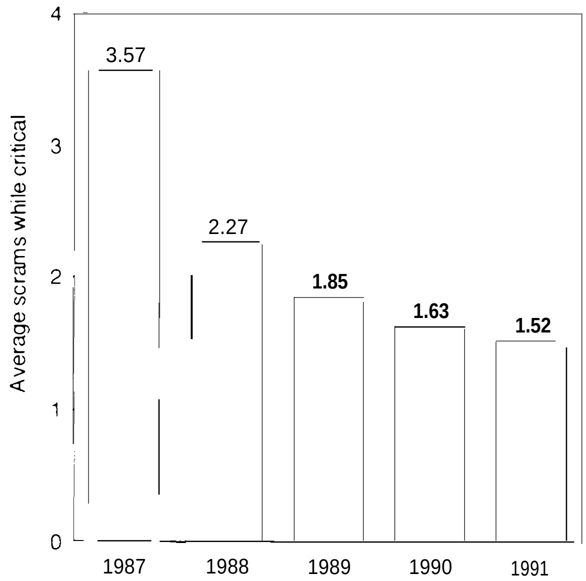
1 U.S. Nuclear Regulatory Commission, *Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Act/v/t/es,* NUREG 1395 draft (Washington, DC: March 1990), p. 13.

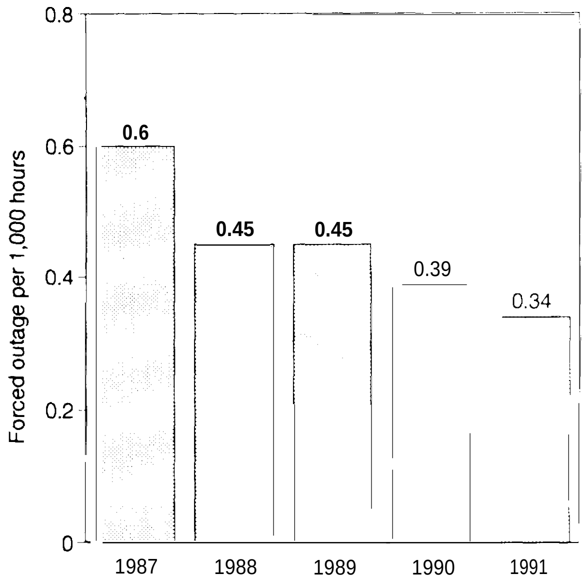
2 Institute of Nuclear Power Operations, "1992 Performance Indicators for the U.S. Nuclear Industry," (Atlanta, GA: March 1993).

Figure 2-4—Average Number of Reactor Scrams While Critical



SOURCE: U.S. Nuclear Regulatory Commission, 1991 *Annual Report*, NUREG-1145, vol. 8 (Washington, DC: July 1992), p. 52.

Figure 2-5--Average Equipment Forced Outage Rate Per 1,000 Critical Hours



SOURCE: U.S. Nuclear Regulatory Commission, *1991 Annual Report*, NUREG-1145, vol. 8 (Washington, DC: July 1992), p. 52.

will be a substantial increase (beyond *adequate* protection) in the overall protection of the public health and safety and if the implementation costs warrant this increased protection. Because license renewal extends a plant operating life, the safety benefits estimated in the backfit analysis will generally be greater than under the original license term. The extent to which potentially costly backfits will be required as a condition of license renewal has not been determined.

HEALTH AND SAFETY GOALS FOR AGING PLANTS

Public Health and Safety Goals for Nuclear Power Plants

To address the issue of acceptable public safety risks from operating nuclear power plants, the NRC set formal, qualitative safety goals for plant operations in 1986 after several years of development.[72] The goals established by the NRC for public and occupational health and safety for existing plants do not change as the plants age. The goals, which apply to existing as well as future plants, are:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The NRC also set the following quantitative objectives for risk of immediate deaths caused by a radiological accident and for deaths from cancer to be used in determining achievement of the goals:

72 51 *Federal Register* 30028 *et seq.* (Aug. 21, 1986).

Box 2-H--Environmental Qualification of Electrical Equipment

A wide variety of electrical cables from different manufacturers are used in nuclear power plants for instrumentation and controls. Cables used in fossil-fuel power plants have generally performed well for as much as 60 years, even though the materials used were inferior& newer cables.[1] Cables used in nuclear plants have a similar excellent operating history. However, aging degradation resulting from high temperature and radiation may go undetected and result In inadequate performance under the additional environmental stresses of accident conditions. Cables required to perform a safety function during and followinga design basis event are required to qualified considering the effects of aging.[2]

Institute of Electrical and Electronics Engineers (IEEE) standards adopted in 1974 and incorporated in NRC requirements specify an environmental qualification (EQ) procedure involving accelerated aging of test samples to ensure that aged cables perform adequately under accident conditions.[3] However, EQ testing of pre-aged samples was not required for the more than SO plants receiving construction permits before June 1974, although consideration of aging effects were to be considered in design. Cable testing and surveillance within a plant's containment is minimal, because they are often hard to access.

The NRC conducts an extensive, ongoing cable testing program at Sandia National Laboratories, which examines a wide variety of cables.[4] The results generally indicate that most popular cable types should perform adequately during current plant operating license and any renewed terms, although there maybe some exceptions requiring further analyses? Similariy, EPRI initiated a multiyear project in 1985 to compare natural and artificial aging for a limited number of cable types.[6] Initial results have found no changes in material properties of concern.

Overall, electrical equipment performance has been excellent, research results on cable aging have been favorable, and EQ has not raised near-term concerns for plant operation, but both the NRC and the commercial nuclear power industry continue to address some longer term issues. NRC staff, for example, recently proposed re-examining the adequacy of current EQ requirements as a generic safety issue.[7] Among the issues that may have long-term impacts are the following:

- the accuracy of EQ methods involving artificial aging,
- the appropriateness of current EQ requirements for cables for which artificial aging tests were not required, and
- a lack of effective testing and surveillance methods to detect degradation.

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1 A.S. Amar, et al., *Residual Life Assessment of Major Light Water Reactor Components—Overview,* NUREG/CR-4731 (Washington, DC: U.S. Nuclear Regulatory Commission, November 1989).

210 CFR 5049.

3 IEEE 383-1974; incorporated in 10 CFR 50.49.

4 Sandia National Laboratories, *Aging, Condition Monitoring, and Loss of Coolant Accident Tests of Class 1E Electrical Cables,* NUREG CR-5772, vol. 1-3 (Washington, DC:Nuclear Regulatory Commission, 1992).

5 For example, Sandia tests recently Identified a potential deficiency for one specific brand of cable when used according to its environmental qualification. A. Thadani, U.S. Nuclear Regulatory Commission, Memorandum to Steven Varga, Director, NRC Division of Reactor Projects, Jan. 27,1993.

6 University of Connecticut, *Natural Versus Artificial Aging of Nuclear Power Plant Components,* EPRI TR-100245 (Palo Alto, CA: Electric Power Research institute, January 1992).

7 U.S. Nuclear Regulatory Commission, SECY 93-049, Mar. 1, 1993.

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes,

These goals provide useful guidance in evaluating the adequacy of plant safety and in developing and implementing regulatory requirements. There remain, however, some limitations to the safety goal policy as it relates to plant aging and to existing plants generally. Limitations to the safety goal policy include the practical translation of risk-based goals into regulatory activities, no consideration of changing population characteristics near a plant, no discussion of the cost-benefit analyses now used in safety decisions, and an unclear relationship and consistency with safety goals found in other Federal law.

Perhaps the greatest weakness of the safety goal policy is the practical difficulty of translating the risk-based goals into regulatory practices. The relationship between many of NRC's regulatory activities and its safety goals is unclear. For example, the safety goal policy is not mentioned in the license renewal rule, the 32-page Statement of Considerations accompanying the rule,[73] or the NRC's regulatory analysis of the rule.[74] Similarly, the most recent plan for the NRC Nuclear Plant Aging Research (NPAR) program does not reference the safety goal policy in any of its approximately 170 pages.[75] One aging-related example of a regulatory effort explicitly incorporating risk issues is the maintenance rule, which requires consideration of risk-significance in the development of maintenance programs.[76] The NRC has an ongoing effort to make greater application of the safety goal policy.[77]

A second limitation with the safety goal policy is indirectly related to plant age: the changing population characteristics over the life of a plant are not addressed. When the safety goal was first adopted, one NRC Commissioner noted that the safety goals do not explicitly include population density considerations; a power plant could be located in Central Park and still meet the standard.[78] Population density and other related demographic characteristics (e.g., transportation facilities) can all change over the decades a plant is in operation.

Regarding the use of cost-benefit analyses, the backfit rule allows the NRC to require safety efforts that surpass those necessary for the adequate protection of public health and safety.[79] These safety efforts must meet an economic test, comparing costs with the expected benefits of improved safety. This suggests a third limitation with the safety goal policy, because it does not address the appropriateness of mandating activities not necessary for adequate protection, or the

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73 56 *Federal Register* 64943-64980 (Dec. 13, 1991).

74 U.S. Nuclear Regulatory Commission *Regulatory Analysis for Final Rule on Nuclear Power Plant License Renewal,* NUREG-1362 (Washington, DC: October 1991).

75 U.S. Nuclear Regulatory Commission, *Nuclear Plant Aging Research (NPAR) Program Plan,* NUREG-1144, Rev. 2 (Washington, DC: June 1991).

76 10 CFR 50.65(a)(3).

77 U.S. Nuclear Regulatory Commission "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy, SECY-91-270.

78 51 *Federal Register* 30033 (Aug. 21, 1986).

79 10 CFR 50.109.

role of economic analyses in supporting those requirements. This can be an important license renewal issue, as the extended operating period results in higher estimated benefits. Specifically, license renewal may result in additional costs for NRC-mandated back.tits not required for adequate safety.

A fourth limitation with the safety goal policy is unrelated to plant aging but relevant to determining the adequacy of the goals: indications of consistency with safety goals found in other Federal law. Nuclear power plants are not unique among electricity supplies in imposing public health and safety risks. Production and use of fossil fuels contribute to health problems ranging from respiratory disease related to particulate and sulfur oxides, to cancers associated with carcinogenic releases from petrochemical facilities, to fatal accidents in the mining and transportation of coal.[80] Heavy use of fossil fuels also produces substantial CO_2 emissions, which contribute to the chance of potentially catastrophic public health and safety impacts resulting from global environmental change. Even energy efficiency measures can create public health and safety risks. For example, better sealed houses can result in indoor air quality problems, such as increased radon exposures. Although the NRC safety goal suggests comparing nuclear plant risks to the risks of other generating sources, a belief that "the absence of authoritative data make it impractical to calibrate nuclear safety goals by comparing them with coal's risks based on what we know today, ' led the NRC to omit quantitative objectives for explicitly assessing that portion of the goal.[81]

The Impact of Aging on the Attainment of Safety Goals

The best available evidence indicates that NRC's public safety goals are met with wide margins, and should continue to be met as plants age, assuming effectively designed and implemented maintenance programs and continuing research to identify latent aging effects. There will always remain some risk and uncertainties, however, and continued nuclear industry and Federal regulatory vigilance remains crucial to implement current practices and to revise them as necessary.

Regardless of plant aging effects, the public cancer risk from normal nuclear plant operation appears very low relative to the NRC goal. Aging management activities, such as equipment replacement and other maintenance work, are primarily contained within a plant.[82] According to the NRC, these activities are unlikely to alter the offsite radiation exposures currently experienced.[83] The estimated public radiation doses from nuclear power plants are extremely low—very far below the allowed maximum.[84] In part, estimated public doses are far below regulatory ceilings, because of an additional regulatory requirement to limit exposures to "as low as is reasonably achievable" (ALARA).[85]

In 1988, the estimated average annual dose for a member of the public residing near a nuclear

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80 U.S. Environmental Protection Agency, Office of Air and Radiation, "Regulatory Impact Analysis on the National Ambient Air Quality Standards for Sulfur Oxides (Sulfur Dioxide)," draft, May 1987, chapters 6 and 7.

81 51 *Federal Register* 30030 (Aug. 21, 1986).

82 Every nuclear plant releases some radionuclides during normal operations to which the public may be exposed. (Some coal Power plants also release some radionuclides, depending on impurities in the coal.)

83 U.S. Nuclear Regulatory Commission, *Environmental Assessment for Final Rule on Nuclear Power Plant License Renewal*, NUREG-1398 (Washington DC: October 1991), p. iii.

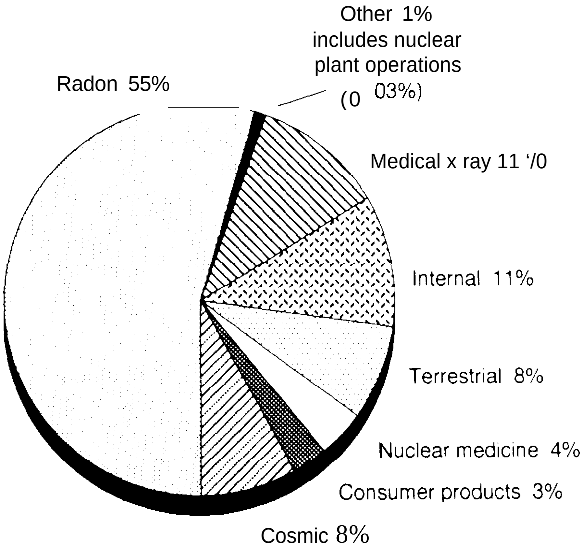
84 Beginning in 1994, the maximum annual exposure limit for a member of the public living near a nuclear plant is lowered from 500 to 100 mrem, still thousands of times higher than estimated maximum exposures. 10 CFR 20. 1301(a).

85 ALARA involves "taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest. ' 10 CFR 20.1(c).

plant was about 0.001 mrem.[86] This dose represents a very small fraction of the total exposure from all sources, including natural ones such as cosmic rays or radon-bearing granite (figure 2-6). The best available evidence indicates that the excess cancer risk to the public from operating nuclear power plants is less than 0.00003 percent, over three orders of magnitude below the safety goal of 0.1 percent additional cancer mortality risk.[87] There are uncertainties in estimating health impacts for any level of radiation exposure (box 2-I). If, however, future exposures and risk remain even remotely similar to past experience, the safety goal should be readily met.

With regard to accident risks, the best available information, although inherently uncertain, indicates that if aging is properly managed the risk of fatalities resulting from a severe nuclear power plant accident in the United States is low relative to the NRC safety objective. For example, the NRC's best and most detailed estimates indicate that an individual near a nuclear plant faces a risk from a plant accident of less than 0.02 per million (figure 2-7).[88] In contrast, the accidental death rate in 1990 from non-nuclear accidents for the U.S. population was about 370 per million people, or over 18,000 times higher.[89] Thus, the NRC's safety objective for prompt fatality risk appears met by at least a factor of about 18.

Figure 2-6—Average Annual Background Radiation Exposure, U.S. Population (360 millirems)



SOURCE: **National Research Council,** ***Health Effects*** *of Exposure to* ***Low Levels of Ionizing Radiation BEIR*** *V* **(Washington, DC: National Academy Press, 1990), p. 19.**

For context, consider the accidents at Three Mile Island (TMI) in 1979 and at Chernobyl in 1986, neither of which was related to power plant aging. At TMI, there were no immediate fatalities, and the best estimate of resulting cancer fatalities over the next several decades is zero. 90 Despite a partial core meltdown, there was no containment

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86 The estimated *maximum* annual dose received by any member of the public in 1988 was 0.02 mrem. D.A. Baker, Pacific Northwest Laboratory, *Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1988,* NUREG/CR-2850, vol. 10 (Washington DC: U.S. Nuclear Regulatory Commission January 1992), pp. iii, 1.4-1.5.1988 is the most recent year for which estimated exposures were readily available from the NRC. Radiation monitoring systems at various locations within and around each plant are used, but radiation levels beyond plant boundaries are often too low to register sufficient information about the exposure of neighboring populations. Therefore, annual exposures for neighboring populations within 50 miles of power plants are estimated based on known releases. Tom Essig, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, personal communication Feb. 18, 1993.

87 For an annual lifetime dose of 100 mrem, the best estimate of excess cancer mortality is about 3 percent. Committee on the Biological Effects of Ionizing Radiations, National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V,* (Washington, DC: National Academy Press, 1990), pp. 172-173. Assuming a linear dose-response relationship (which is necessarily uncertain), an annual average exposure of 0.001 mrem then would produce a risk of excess cancer mortality of 0.00003 percent. If actual exposures approached the maximum exposure limit rather than ALARA, the best available information indicates that NRC's safety goal would *not be* met.

88 U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear power Plants,* NUREG-1 150, vol. 1 (Washington, DC: December 1990), p. 12-3.

89 U.S. Bureau of the Census, Statistical Abstract *of the United States: 1992,* 112th ed. (Washington, DC: 1992), p. 82.

90 J.I. Fabrikant, "Health Effects of the Nuclear Accident at Three Mile Island," *Health Physics,* vol. 40, February 1981, pp. 155-156.

Box 2-1-Estimating Health Impacts From Public Radiation Exposure

There are two principal approaches to determining the public health impacts of normal nuclear power plant operations: 1)epidemiological studies comparing the health of populations living near plants to other populations, and 2) risk assessment, which involves estimating accident probabilities and their consequences in order to calculate exposure levels and health impacts.

Several epidemiological studies of public exposure from nuclear power plants and their health impacts have been performed, but results have varied. Some studies found increased cancer incidence, while others actually found decreased incidences.[1] At present, there are no national data that indicate that current public exposures to radiation from operating power plants produce detectable increases in cancer deaths.[2] Epidemiological studies of radiation cancer risks from nuclear power plants rarely, if ever, have enough information to provide complete or conclusive results, because the risk is generally too low to measure and data needs can be substantial. For example, researchers performing epidemiological studies must identify appropriate control populations, follow or obtain data on the status of exposed populations overlong periods (generally decades), and obtain reliable data on cancer incidences and deaths from both study and control populations. Gathering such information over wide geographic areas is extremely difficult and requires an extended research commitment, in terms of both time and funds. In addition, cancer caused by radiation cannot generally be distinguished from cancer caused by other sources. This complicates efforts to identify sources of risk when there are detectable cancer increases in a study population exposed to low levels of radiation.

Furthermore, as many epidemiological studies of populations exposed to the very low levels of radiation associated with operating nuclear power plants have been inconclusive, current estimates of the radiation health impacts of low doses are generally based on data from high exposures-such as the atomic fallout from the 1945 bombing of Japan. These data are generally extrapolated linearly to estimate the risks of lower radiation doses, such as those experienced by residents near nuclear power plants. However, there are substantial uncertainties in extrapolating risk estimates from high doses to low doses.[3] In particular, risk may not have a linear relationship relative to dose but may, in fact, decrease below a certain dose threshold. On the other hand, the opposite may be true, and risks are likely to vary depending on other factors such as the age and health of a population. Thus, risk assessments based on linear dose-response relationships remain inherently uncertain.

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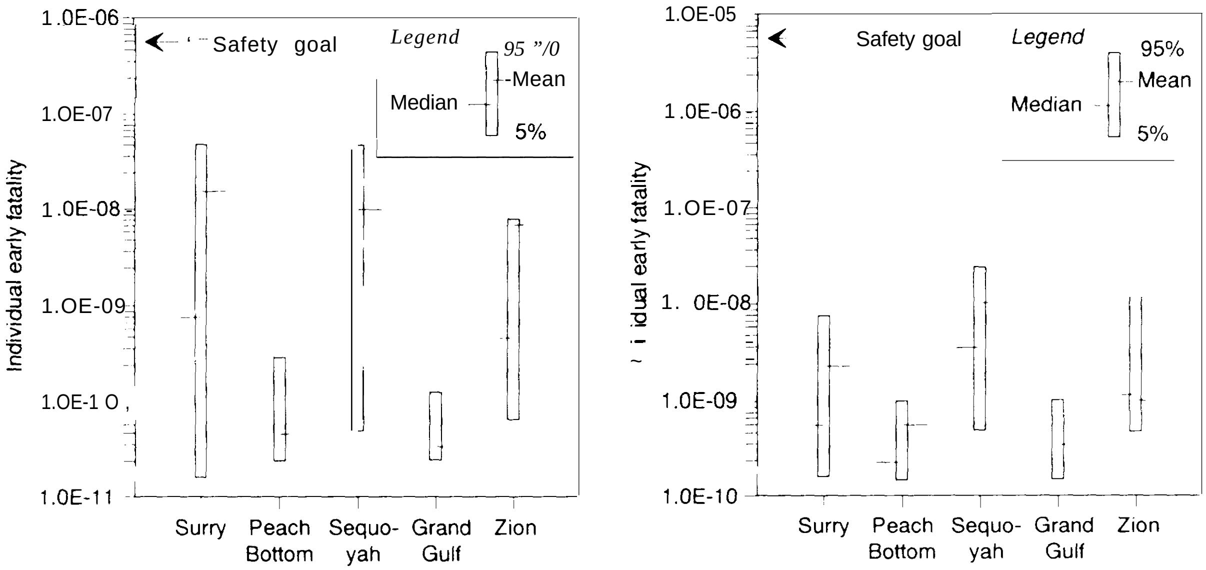
1 See, for example, committee on the Biological Effects of ionizing Radiations, National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V* (Washington, DC: National Academy Press, 1990), pp. 377-379; and S. Jablon, Z. Hrubec, J.D. Boice, Jr., and B.J. Stone, National Cancer Institute, *Cancer in Populations Living Near Nuclear Facilities* (Washington, DC: U.S. Government Printing Office, July 1990), vol. 1, Report and Summary, pp. 8-15.

2 In this context, a "detectable" increase is one that can be distinguished from the expected number of cases in a population. See S. Jablon, Z. Hrubec, and J.D. Boice, Jr., "Cancer In Populations Living Near Nuclear Facilities: A Survey of Mortality Nationwide and Incidence In Two States," *The Journal of the American Medical Association*, Mar. 20,1991, vet. 265, No. 11, pp. 1403-1408.

3 See, for example, Committee on the Biological Effects of Ionizing Radiations, Nationat Research Council, *Health Effects of Exposure to LowLevels of Ionizing Radiation: BE/I? V* (Washington, DC: National Academy Press, 1990), pp. 1-8. As explained in this source, risk projections for solid tumors are linear, while those for leukemias are linear quadratic.

Figure 2-7-Comparison of Probabilistic Risk Assessment Results With Safety Goals (per reactor year)



SOURCE: U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,* NUREG-1140, vol. 1 (Washington, DC: December 1990), p. 12-1.

breach at TMI, and the radiation released was low. People living within 10 miles of the plant, who experienced the highest estimated exposures, received an estimated average 6.5 milliems, a small fraction of the annual background radiation level. No radiation levels above background were detected beyond the 10-mile radius of the plant.

In contrast, the 1986 Chernobyl accident caused widespread release of large amounts of radionuclides and caused about 30 prompt fatalities--most of them emergency workers. The best estimate of resulting cancer fatalities is about 17,000, or about 0.01 percent above the background cancer fatality rate expected over the remaining lifetimes for the affected European population. The health risk to the population living near the plant is much greater. About 24,000 of the 115,000 people evacuated from the surrounding area received an average of 43 reins. This dose is estimated to lead to an additional 26 fatal leukemias over their lifetimes, a risk increase of 200 percent for a group this size.[91]

The public risk from a nuclear accident depends on two factors: 1) the probability of a severe accident with a substantial offsite release of radiation, and 2) the consequences on the exposed population, Unmitigated aging degradation, or other factors that change over time, can affect both the probability of an accident and the severity of the consequences. For example, the probability of an accident involving a large release of radionuclides depends on the frequency of initiating events (e.g., human errors, equipment failures, loss of offsite power) and the subsequent events that might lead to reactor core and containment damage. Inadequately managed, aging degradation can increase the probability of equipment failure, thereby affecting both initiating events and the ability to manage an accident.

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[91] M. Goldman, R. Catlin, and L. Anspaugh, *Health and Environmental Consequences of the Chernobyl Nuclear Power Plant Accident*, DOE/ER-0332 (Washington DC: Office of Energy Research, U.S. Department of Energy, June 1987), pp. vii-xv.

Offsite conditions may also change over time, such as changing population settlement patterns around a plant, and thus alter the potential consequences of an accident.

For decades, the NRC and the commercial nuclear power industry have worked to understand better and quantify public accident risks. In 1975, the NRC completed a much criticized study of the probabilities and consequences of severe accidents at two commercial nuclear facilities using PRA techniques for the first time.[92] Following the TMI accident, the NRC commissioned indepth PRAs of five nuclear plants representing major U.S. reactor designs (Zion, Surry, Sequoyah, Peach Bottom, and Grand Gulf).[93] For these "reference plants," the NRC estimated mathematical probabilities of complex system failures and public health consequences. As estimated in that effort, the risks are at least one, and perhaps as many as five, orders of magnitude below the current NRC safety goal. The reference plant study did not explicitly address aging and assumed that aging management programs were sufficient to maintain current equipment performance.

Because small differences among otherwise similar plants can create significant differences in risk, the NRC in 1988 required all utilities to conduct probabilistic studies of their own plants called individual plant examinations (IPEs).[94] IPE results were intended to improve the understanding of the types of severe accidents possible at each plant and to ensure that no undetected, plant-specific accident vulnerabilities existed. Utilities are required to develop accident management methods for identified vulnerabilities.[95]

PRAs are subject to substantial uncertainties. Commenting on the NRC PRA study of five nuclear power plants, the Advisory Committee on Reactor Safeguards (ACRS) noted that the "results should be used only by those who have a thorough understanding of its limitations."[96] These limitations include the following:

- limited historical information regarding the failure rates of critical equipment, particularly from aging effects;
- the difficulty of modeling human performance (e.g., the behavior of plant operators before and during an accident); and
- the lack of information regarding containment performance.

The cost of performing PRA analyses can be substantial; the NRC estimated that the IPE program would cost operators an average of between $1.5 million to $3 million per plant. Despite their limitations, PRA methods can be useful to identify risks and set priorities for additional research and analysis. Utilities have applied PRA methods and results to a variety of operations, maintenance, and economic decisions.[97]

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92 U.S. Nuclear Regulatory Commission, *Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,* WASH-1400, NUREG-75-014 (Washington, DC: October 1975); and U.S. Congress, Office of Technology Assessment, *Nuclear Power in an Age of Uncertainty,* OTA-E-216, (Washington DC: U.S. Government Printing Office, February 1984), pp. 218-219. The NRC study was initiated by its predecessor agency, the AEC.

93 U.S. Nuclear Regulatory Commission, Severe *Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,* NUREG-1 150, vol. 1 (Washington DC: December 1990). That analysis is reviewed in American Nuclear Society, "Report of the Special Committee on NUREG-1 150, The NRC's Study of Severe Accident Risks," June 1990.

94 D.M. Crutchfield, "Individual Plant Examination for Severe Accident Vulnerabilities," U.S. Nuclear Regulatory Commission, Generic Letter 88-20, Nov. 23, 1988; and U.S. Nuclear Regulatory Commission, *Individual Plant Examination: Submittal Guidance,* NUREG-1335 (Washington, DC: August 1989).

95 U.S. Nuclear Regulatory Commission, "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

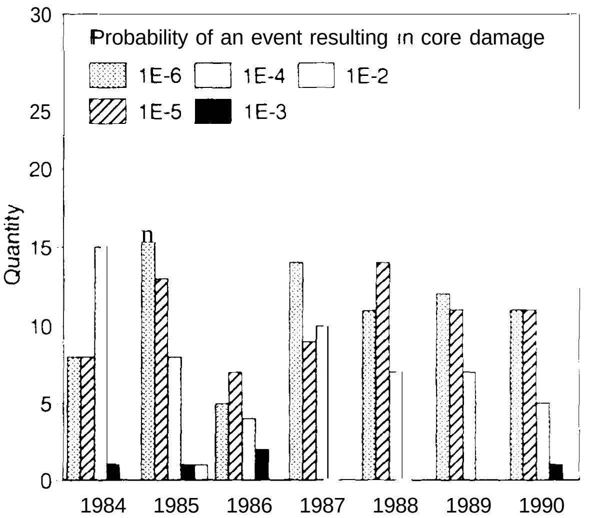
96 U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, Letter to NRC Chairman Kenneth M. Carr, Subject: Review of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Nov. 15, 1990,

97 Yankee Atomic Electric Co., *Applications of PRA,* EPRI NP-7315 (Palo Alto, CA: Electric Power Research Institute, May 1991).

To address aging issues more directly, the NRC NPAR program works to incorporate aging information into PRAs. Age-dependent PRAs model the effects of maintenance practices and the effects of aging on component failure rates, which standard PRAs assume are constant. These studies indicate that aging can have a substantial impact on reactor core damage if maintenance programs are inadequate.[98] However, age-dependent PRAs lack sufficient data to determine accurately aging effects on component failure rates and the effectiveness of different maintenance practices. As a result, they remain an area for continued analysis.[99]

Although accidents involving severe core damage are expected to be extremely rare (e.g., less than once per hundred years in the United States), there are actual operational experiences that can complement PRA results. In particular, NRC's Accident Sequence Precursor program tracks abnormal operating events[100] that could potentially lead to severe accidents.[101] The program uses PRA techniques to determine the significance of those events in terms of the likelihood of core damage. In 1990, 28 operational events were identified as resulting in probabilities of subsequent severe core damage of greater than one in one million. The worst six of those events were estimated to have core damage probabilities of between 1 in 1,000 to 1 in 10,000 (figure 2-8).[102] That is less, by a factor of between 1.7 and 18,

Figure 2-8—Accident Sequence Precursor Quantities, 1984-1990



SOURCE: U. S. Nuclear Regulatory Commission, *1991 Annual Report,* NUREG-1145, vol. 8 (Washington, DC: July 1992), p. 52.

than would be expected based on a core melt frequency of one per 10,000 years per plant.

Occupational Health Impacts

Nuclear power plant workers are generally exposed to more radiation than the residents neighboring their respective plants. Whereas the average member of the U.S. population is annually exposed to an effective total dose of 360 millirems (0.36 reins) from all sources,[103] current

98 Science Applications International Corporation, *Evaluations of Core MeltFrequencyEffects Due to ComponentAging and Maintenance,* NUREG/CR-5510 (Washington, DC: U.S. Nuclear Regulatory Commission, June 1990); and U.S. Nuclear Regulatory Commission, *Regulatory Analysis for Final Rule on Nuclear Power Plant License Renewal,* NUREG-1362 (Washington DC: October 1991), appendix C.

99 Science Applications International Corporation, *Approaches for Age-Dependent Probabilistic Safety Assessments with Emphasis on Prioritization and Sensitivity Studies,* NUREG/CR-5587 (Washington, *DC:* U.S. Nuclear Regulatory Commission, August 1992); and A.P. Donnell, Jr., Sandia National Laboratories, "A Review of Efforts to Determine the Effect of Age-Related Degradation on Risk," SAND91-7093, February 1992.

100 Under 10 CFR 50.73, licensees must submit a Licensee Event Report (LER) when preestablished limits are exceeded or certain events occur. These reports serve as a primary source of operational event data. The threshold for reporting considers infrequent events of significance to plant and public safety as well as more frequent events of lesser significance that are more conducive to statistical analysis and trending.

101 Oak Ridge National Laboratory, *Precursors t. Potential Severe Core Damage Accidents: 1990 A Status Report,* -G/CR-4674 (Washington DC: U.S. Nuclear Regulatory Commission August 1991).

102 U.S. Nuclear Regulatory Commission, *Annual Report 1991,* NUREG-1145, vol. 8 (Washington, DC: July 1992), p. 54.

103 Committee on the Biological Effects of Ionizing Radiations, National Research Council, *Health Effects of Exposure to Low Levels Of Ionizing Radiation: BEIR V,* (Washington DC: National Academy Press, 1990), pp. 18-19.

NRC regulations allow nuclear plant workers to receive as much as 3,000 millirems (3 reins) per calender quarter up to a limit of 5,000 millirems (5 reins) per year, although ALARA goals encourage lower exposures.[104] The average annual measurable added radiation exposure for U.S. nuclear plant workers in recent years has been about 400 millirems.[105] Individual exposures vary, but few exceed the 5-rem limit. Between 1985 and 1989, only two of the approximately 210,000 monitored nuclear power plant workers experienced doses exceeding 5 reins.[106]

Increased maintenance activities associated with aging can increase occupational exposures. More frequent monitoring and testing of SSCs can lead workers to spend additional time in areas with concentrations of radionuclides. Major repairs also lead to additional exposures. For example, the additional collective exposures resulting from replacing a steam generator has been several hundred person-reins, the same order of magnitude as typical annual plant exposures otherwise occurring. However, for those plants requiring them, steam generator replacements are expected only once or twice over the life of a plant.

In 1991, the International Commission on Radiological Protection (ICRP), an international body established in 1928 to develop guidelines for radiological health protection, recommended reducing the accepted levels of occupational radiation exposures from 5 reins per year to 2 reins per year, when averaged over a 5-year period (i.e., a total maximum of 10 reins over a 5-year period). The recommendation to limit the maximum occupational exposure in any single year to 5 reins was retained.[107] Although the NRC generally follows ICRP recommendations, an NRC decision to comply with the 1991 ICRP recommendation was postponed. As part of that decision, the NRC cited recently reduced U.S. occupational exposures from ALARA efforts to levels that already approximate the recent ICRP recommendations.[108]

Although occupational radiation exposures are carefully monitored, determiningg some of the incremental health risks to workers is difficult. For example, epidemiological studies of cancer risk lack reliable data on the risks of whole body radiation exposures below 10 reins (i.e., 10,000 millirems).[109] Nonetheless, the risk models in the BEIR V report estimate that a working lifetime exposure of 1,000 millirems annually (i.e., 1 rem per year each year between the ages of 18 and 65, or one-fifth the allowed maximum) will lead to an increased cancer mortality rate of roughly 15 percent above expected levels.[110]

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104 10 CFR 20.101. In a recent rulemaking, the NRC decided to drop the quarterly limit. *56 Federal Register 23368, 23396* (May 21, 1991). *This rule will be effective in 1994.57 Federal Register 38588* (Aug. 26, 1992).

105 C.T. Raddatz and D. Hagemeyer, *Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities: 1989,* Twenty Second Annual Report, NUREG-0713, vol. 11 (Washington, DC: U.S. Nuclear Regulatory Commission, April 1992), pp. 3-4. Note: The average *measurable exposure* differs from the average individual exposure, because not all nuclear plant workers show measurable exposures. If all workers are considered, the average individual dose for commercial nuclear plant workers is much lower (about 200 millirems in recent years).

106 Ibid., p. 5-10. Because 10 CFR Part 20 rules have allowed annual averaging, an individual exposure greater than 5 rems in any year was not automatically a violation, as long as the age-adjusted annual average remained at 5 rems or less. Under new rules taking effect in 1994, such averaging is no longer allowed, and 5 reins will be the maximum limit for each year.

107 International Commission on Radiological Protection, 1990 *Recommendations of the International Commission on Radiological Protection,* ICRP Publication 60 (New York: Pergamon Press, 1991), pp. 72-73.

108 56 *Federal Register* 23360, 23363 (May 21, 1991).

109 J.I. Fabrikant, "Health Effects of the Nuclear Accident at Three Mile Island," *Health* Physics, February 1981, vol. 40, p. 153. Note: This source actually notes 10 rads, but the units generally convert directly to reins on a 1:1 basis when considering gamma exposures, the exposure of concern with commercial nuclear power.

110 Committee on the Biological Effects of Ionizing Radiations, *National* Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V,* (Washington, DC: National Academy Press, 1990), pp. 172-173.

According to one source, the 107,019 workers exposed to the average 410 millirems in 1988 will experience a risk of additional cancer deaths of 0.2 percent (two cases per thousand); the single individual exposed to 6,100 millirems that year will experience an additional cancer mortality risk of 0.4 percent (four chances in one thousand).[111] As discussed earlier, however, there are many uncertainties associated with such estimates, particularly assumptions about the validity of transferring the results of high-dose exposures to low-level exposures.

The comparative occupational health risks between nuclear power and other energy sources, particularly coal, may be worth examining in more detail. Understanding these comparative risks is important in evaluating the comparative risk-benefits of any energy source. Although all health effects, particularly deaths, are important, there are data that indicate the comparative occupational health risks associated with nuclear power are low relative to other energy sources. For example, the number of deaths and occupational injuries associated with coal production may be far higher than nuclear energy production.[112] OTA has not evaluated such claims for this report, but evaluating the merits of commercial nuclear power plant life attainment and license renewal requires a recognition, if not a complete understanding, of these comparative risks.

111 C.T. Raddatz and D. Hagemeyer, *Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities: 1988*, Twenty First Annual Report, NUREG-0713, vol. 10 (Washington, DC: U.S. Nuclear Regulatory Commission July 1991), p. 4-29.

112 J.I. Fabrikant, "Is Nuclear Energy An Unacceptable Hazard to Health?" *Health Physics*, September 1983, vol. 45, No. 3, p. 576.