3. R&D FOR MANAGING PLANT AGING

Degradation of materials in reactor plant structures and components caused by radiation, high temperatures, high pressure cyclic stresses, and a relatively corrosive environment costs utilities that operate light water reactors (LWRs) hundreds of millions of dollars each year. Not only does this degradation increase current operating costs significantly, but, if continued, material degradation could force the premature shutdown of some plants before the end of their initial 40-year license term. Even for plants that can operate until the end of their current license period, material degradation issues will strongly affect license renewal decisions.

For example, one pressurized water reactor (PWR) plant (Trojan) has already shut down in part because of the cost of replacing its steam generators (SG), which were damaged by stress corrosion cracking (SCC). SG replacements are expensive, on the order of $150 million plus the cost of replacement power. Over fifteen PWR plants in the U.S. have already replaced their original SGs, but even for those that do not yet require replacement, repair and inspection are significant economic burdens. R&D may lead to solutions or mitigators of this phenomenon. In addition, extensive cracking has occurred in the core shrouds that surround many of the boiling water reactor (BWR) cores. Mechanical clamping systems have been devised to repair these components, but SCC of reactor internals exposed to high radiation (irradiation-assisted SCC – IASCC) continues to be an expensive problem for BWRs. Also, the irradiation of LWR reactor pressure vessels (RPV) that occurs during reactor operation reduces the ductility of the vessels. Such irradiation embrittlement leads to operational restrictions that can adversely affect the efficiency and life of a plant and its ability to remain economically competitive. In one case, a decision to permanently shut down a reactor (Yankee Rowe) was strongly influenced by the RPV embrittlement and its impact on plant economics. The programs that monitor the degree of embrittlement of the RPVs at all nuclear plants in order to ensure that they have adequate ductility are expensive.

Material degradation management for operating reactors requires the ability to assess the condition of the materials and hence the operability of the affected structures and components. The accuracy and reliability of these assessments control the degree of conservatism that must be assumed in assessing the effects of the degradation and resultant costs. For example, some types of SG tubing defects can be characterized using current nondestructive examination (NDE) techniques and NRC permits continued operation with these safe, characterized defects. When defects cannot be characterized as safe (e.g., by NDE or evaluation limitations), the only two options (both expensive) are to remove affected tubes from service by plugging or repair; for example, by sleeving. Similarly, no NDE technique is available to confidently assess the degree of embrittlement of RPVs directly. Assessments currently rely on empirically developed generic correlations that can be highly conservative when applied to specific plants. Even with these conservative analyses, it appears that only a few U.S. reactor pressure vessels may require annealing to mitigate the effects of radiation induced embrittlement in order to reach the end of their current license periods. Overly conservative analyses of the degree of embrittlement could bias decisions on the relicensing of plants and reduce capacity factors and revenue because embrittlement eventually imposes restrictions on the rates at which RPVs can be heated up and cooled down.

Research is needed to understand, characterize, and manage service induced degradation of RPVs, reactor internals, SG tubes, primary system piping, electric cables, and safety-related structures. Technology development needs to be focused on timely detection, mitigation, and prevention of significant long-term effects of aging such as stress corrosion cracking, irradiation assisted stress corrosion cracking, reduction in fracture toughness due to neutron irradiation, thermal embrittlement of cast austenitic stainless steels, piping fatigue, and structure degradation. A research program to address environmental degradation of reactor components, systems, and structures would be a multi-year program involving laboratory tests, component inspections, and technology demonstrations.

This plan element would complement existing industry and government R&D activities, and would be conducted in close coordination with both. Industry-sponsored work is heavily focused on short-term solutions. The ongoing confirmatory research by NRC focuses on issues related to safety. NRC must be assured that assessments of degraded components by the industry are sufficiently conservative to provide adequate safety margins. Said another way, NRC research seeks to characterize the magnitude of the uncertainties involved in the assessment of degraded components, but is not responsible for directly developing technology to reduce these uncertainties. Tight budgets and restricted missions permit relatively little development of base technologies by NRC. For example, the work sponsored by NRC on the NDE of degraded steam generators focuses on the assessment of the capabilities of the current NDE technologies. The development of improved NDE technologies, described in this plan element, is outside the scope of the NRC research responsibilities. Because of its unique facilities and capabilities, DOE is in the best position to address many of the difficult technology issues that involve long-term, high-risk issues. Industry would complement DOE capabilities with its knowledge of current plant issues and assist in developing solutions to both short and long-term issues of material degradation.

The research technology areas discussed below are organized by major reactor component. For each major critical component the plan (1) identifies the relevant aging degradation mechanisms and discusses the issues associated with continued safe and economic operation of that component; (2) summarizes the current research and development activities (which are primarily funded by EPRI and the various owners groups); and (3) identifies the research needs. Volume III includes descriptions of the tasks which have been identified to address the research needs. Volume II provides more detailed descriptions of those projects which are considered to be the highest priority, requiring attention in the near-term. Major critical components discussed are reactor pressure vessels; reactor pressure vessel internals; PWR steam generators; reactor coolant system piping, pumps, and valves; electric cables; and structures. Research needs that are not component-specific are also discussed.

3.1 Steam Generators

Steam generators have historically been among the most troublesome of the major components in commercial PWR plants around the world. As of December 1994, twenty-two plants have already replaced their original steam generators. Such replacements cost on the order of $150 million. Even for those plants that have not yet had to replace their steam generators, inspection, monitoring, and repair are very expensive, costing the nuclear utilities several hundred millions of dollars each year. Degradation of steam generator tubes has resulted from corrosion and wastage, pitting, denting, SCC, fretting (wear), fatigue, and intergranular attack (IGA). A typical PWR steam generator is shown in Figure 3.1. Locations in steam generator tubes affected by these degradation mechanisms are shown in Figure 3-2. The percent of total number of tubes plugged caused by each of these mechanisms for the years 1973 to 1994 is shown in Figure 3.3. Industry efforts have been largely successful in managing degradation due to wastage, pitting, and denting, but fretting, SCC and IGA have proved more difficult to manage. SCC and IGA are most prevalent in regions where the local chemistries may differ significantly from those in the bulk such as crevices at the tube sheet and tube support plates, under sludge piles or other deposits, or in regions of high strain and residual stresses such as the roll transition at the tube sheet.

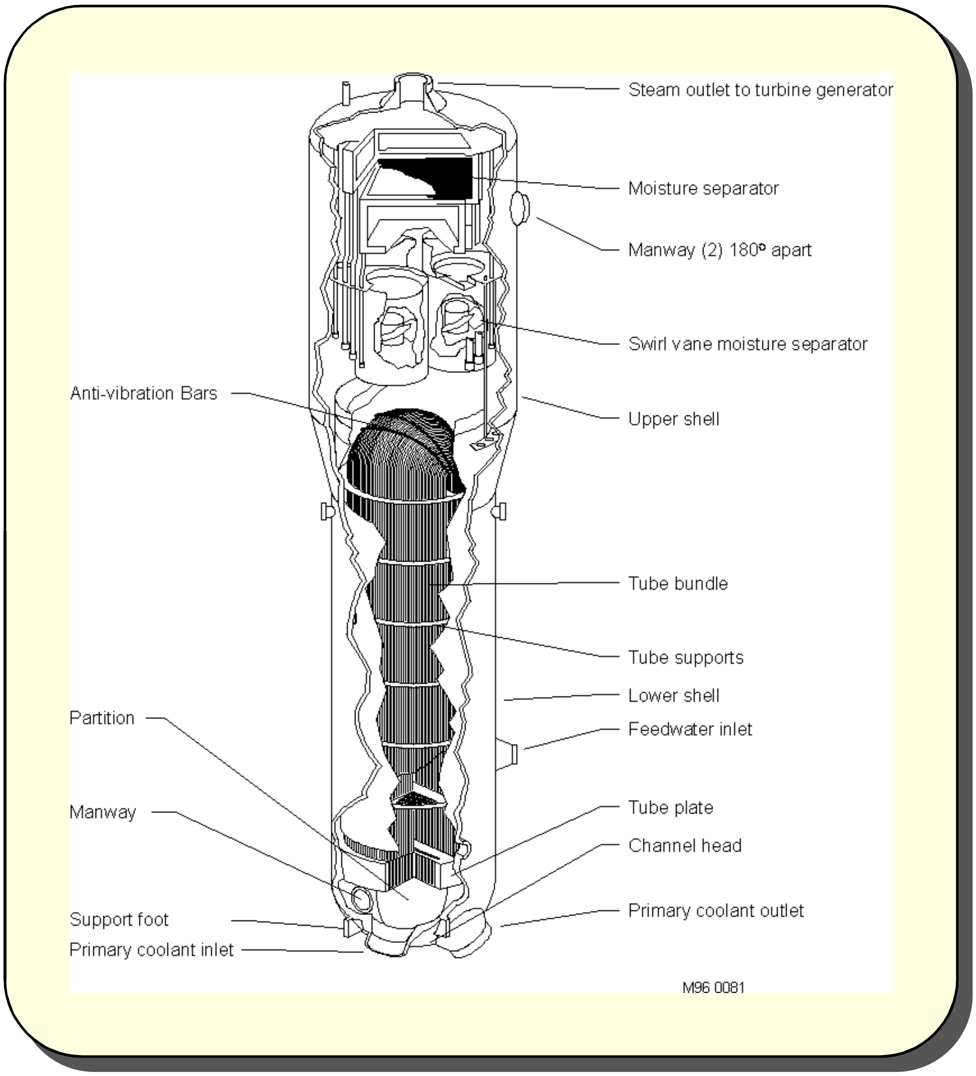

Figure 3-1. Typical PWR steam generator

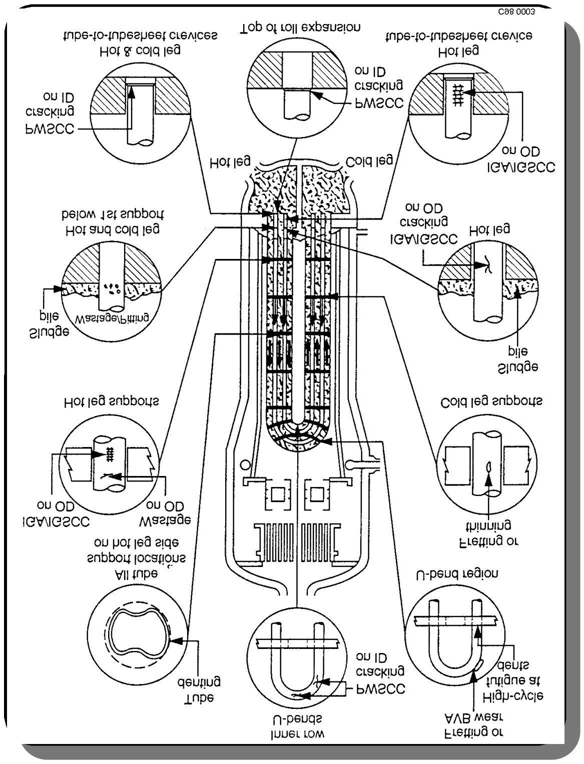

Figure 3-2. Locations of known tube wall degradation in recirculating steam generators

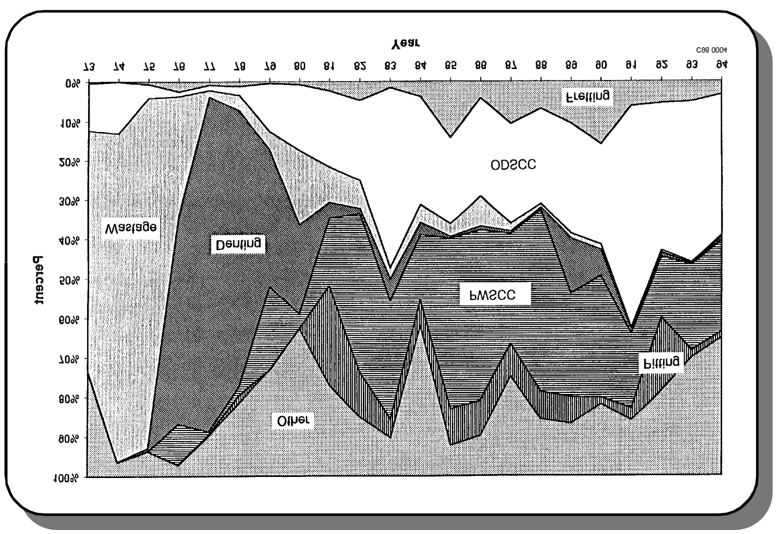

Figure 3-3. Percent of total steam generator tubes plugged due to various degradation mechanisms, 1973 through 1994

Effective management of the consequences of tube cracking requires (1) nondestructive evaluation techniques to characterize the crack, (2) the ability to characterize the impact of the degradation on the structural integrity of the tubing, and (3) the ability to adequately project the rate of degradation so that the integrity of the tubing can be assured throughout the operating cycle until the next inspection. The accuracy and reliability of the NDE control the degree of conservatism that must be assumed in assessing the effects of the degradation. This strongly affects decisions to repair or replace steam generators. For example, for some types of cracking in steam generator tubing, current NDE techniques provide a sufficiently accurate characterization of the cracking that the NRC permits continued operation of a steam generator in which such cracking has been detected (subject to periodic inspection). For other types of cracking for which no NDE technique has been demonstrated to be adequate to characterize the degradation, the only recourse is to plug or repair the tube upon detection of the crack. About 10,000 to 12,000 steam generator tubes are plugged or repaired each year, and based on the results of tube-pulls and in situ pressure tests, the vast majority of them probably have substantial remaining margin against structural failure.

If a crack is detected and it cannot be adequately characterized or if the progression of the degradation is too rapid or unknown, the cracked tube must be repaired or plugged. Current repair technologies (sleeving) are very expensive and have not always performed reliably in the past. Plugging is costly and too much plugging degrades the thermal performance of the plant.

The fundamental cause of many of the corrosion problems encountered in steam generators is that the Alloy 600 material used for tubing was not as corrosion resistant as originally thought. The concentration or hideout of bulk-water, low-concentration, low-volatile ionic impurities in local crevice regions, aggravated the problem. Work to mitigate this type of degradation in current steam generators (i.e., slowing the rate of initiation or progression of the degradation) focuses on understanding and reducing the aggressiveness of the chemistry in crevices. Industry programs to understand the effect of bulk water chemistry on forms of degradation such as wastage and denting have been very successful. However, the relation between bulk water chemistry and the local crevice chemistry is extremely complex (estimates of the concentration factors in crevices range from 40,000 to 1,000,000), and mitigation of stress corrosion cracking by control of the bulk water chemistry has been less successful to date.

3.1.1 Current Research on Steam Generator Reliability

Both EPRI and NRC are sponsoring research on steam generator degradation. EPRI work focuses on improved methods of characterizing the current condition of installed steam generators and of predicting and controlling their future rate of degradation to minimize future costs. Results include new and improved technology aimed at residual life management and current steam generator reliability. Results also include collective industry knowledge which can be used in the design, fabrication, installation, and operation of replacement, new, and advanced LWR steam generators. The NRC program emphasizes the assessment of current and emerging NDE technology and the assessment of tube integrity.

Improved In-service Inspection and NDE. Current work provides for improved steam generator tubing ISI/NDE performance and the capability to detect, size, and characterize the various forms of mechanical and corrosion damage to steam generator tubes. This includes evaluating, adapting and integrating various technologies, enhancing existing or current methodologies, and developing new techniques. Also included is the development and maintenance of detailed guidelines on inspection management. Among these are recommendations for inspection sampling and expansion, inspection frequency, requirements for eddy current data analyst and procedure qualification and performance demonstration (PD), and maintenance and updating of the software packages for data analyst training and qualification. One focus will be on assessing the performance of new improved tubing materials used in replacement steam generators.

<u>Improved Steam Generator Materials Performance</u>. Current efforts build on past work funded through the various EPRI-managed steam generator programs since 1977 to develop an understanding of causes and remedies for the materials-stress-environment interactions that have resulted in substantial PWR performance losses due to steam generator corrosion damage. Current efforts also include addressing potential corrosion mechanisms that could affect newer tubing materials (i.e., Alloy 690), and the potential for secondary side material degradation. Deliverables include development of candidate corrosion inhibitors and buffers, development and verification of flaw initiation and growth models, specifications for improved materials and fabrication features, guidance on and specific data relative to tube-pull analyses, documentation on the occurrence and potential causes of specific damage mechanisms, and guidance on repair methods that impact materials performance (e.g., sleeving).

<u>Improved and Advanced Secondary Water Chemistry Guidelines</u>. The goal is a comprehensive set of guidelines, water treatment schemes, software analysis packages, hardware, and processes to control the environmental conditions on the secondary side of PWR steam generators. Deliverables include input to the PWR secondary water chemistry guidelines, input to and development of supporting application guidelines, alternative feedwater pH control additives, secondary cycle chemistry management and analysis software, and sludge removal processes and hardware.

<u>Steam Generator Defect Specific Management (SGDSM)</u>. The objective of the current work is to develop a methodology for and support specific utilities in establishing tube burst and leakage correlations necessary to justify damage-specific tube plugging and repair limits. Deliverables include damage-specific NDE techniques, data analysis and correlation development, and development and maintenance of the SGDSM database.

<u>Steam Generator Technology Support: Status and History Databases</u>. The current effort provides for industry-wide steam generator performance tracking and activities related to information exchange and technology transfer among utilities. Deliverables and milestones include a database and annual progress report on the status of worldwide steam generator performance and degradation progression, revision to the "Steam Generator Reference Book," and topical workshops and information exchange meetings.

<u>SG Tube Thermal Hydraulics, Vibration and Fatigue Workstation</u>. Past work funded through the various EPRI-managed steam generator programs since 1977 has been focused on developing and validating thermal-hydraulic and flow-induced vibration codes. These codes provide for an improved understanding of the operating in-bundle conditions and are used to predict areas of

susceptibility to fouling and vibration damage, to analyze and understand identified T-H, vibration and fatigue problems, and to improve the T-H and vibration design of new and replacement SGs. The results of the current work will include validated codes, improvements to the codes, workstation software and support of utilities in analyzing T-H, fouling, and vibration problems.

3.1.2 Additional Steam Generator R&D Needs

<u>Improved NDE Techniques</u>. Improvements in the capability of NDE techniques to characterize defects in terms of their impact on structural integrity are critically needed. Development of advanced NDE systems can draw upon the expertise and experience of the DOE national laboratories, universities, and commercial firms. A variety of approaches to the development of advanced NDE techniques for steam generator tubing can be envisioned. Several promising examples are described in the Appendix. Because there is no clear way *a priori* to identify the best approach to developing improved NDE systems, even though EPRI has ongoing programs, the effort by DOE will be complementary, not duplicative. NRC also sponsors work in this area, but the emphasis of its effort is to assess the capability of the NDE being used by industry to characterize degraded tubing rather than to develop improved NDE technologies.

Figure 3-4. McGuire Unit 1 steam generator support plate and tubing

A major problem in the development of such techniques is the availability of suitable test specimens. Most conventional methods for simulating defects such as electro-discharge machining or even laser cutting techniques produce defects that are poor simulations of the stress corrosion cracks produced in the field. Better, but still not completely representative, cracks can be produced in the laboratory, but these are still very expensive and can be produced only in a few locations that have the necessary experience and facilities. A significant portion of the effort under Task 3.7.1, *Assessment of Aging Effects on Components and Structures from Nuclear Power Plants*, will be to obtain additional degraded tubing from actual steam generators, similar to the steam generator support plate and tubing material shown in Figure 3.4 from McGuire Unit 1. This tubing, along with laboratory cracked tubing and other simulated degraded tubing, needs to be maintained in a central facility accessible to all interested parties who wish to evaluate techniques for the detection and characterization of degraded tubing. Where feasible the degraded tubing should be placed in a setting where the artifacts associated with real inspections can be simulated as realistically as possible. After the tubing has been characterized by all the NDE techniques of interest, it should be characterized destructively either by metallographic techniques or by leak or burst tests under near prototypical conditions. The DOE laboratory test facility shown in Figure 3-5 can test degraded tubing under simulated primary and secondary conditions ranging from normal operation to those encountered in a main steam line break. Both axial and circumferential cracks can be tested. The entire progression of the failure from the initiation of a small leak, stable crack growth under increasing pressure, to unstable crack growth, can be simulated in the facility.

Figure 3-5. Argonne National Laboratory test facility for crack initiation and growth testing under simulated reactor operating conditions

Crevice Chemistry and Corrosion. Additional research is needed to better understand the relationship of the bulk water chemistry to the local crevice chemistry. The effect of critical parameters such as bulk water chemistry, degree of crevice superheat, and crevice geometry on crevice chemistry needs to be determined. Instrumentation capable of measuring crevice chemistry conditions at LWR pressures and temperatures needs to be developed. Online corrosion monitoring systems such as electrochemical noise monitoring that may be suitable for field implementation may provide an alternate approach to assessing crevice chemistry conditions.

Improved Repair Technologies. Current repair technologies are expensive and not always effective. Lower cost, highly reliable repair technologies for steam generator tubes are needed.

Improved Assessment of the Consequences of Tube Leakage or Failure. In assessing the consequences of steam generator tube leaks or failure in terms of the exposure of the public to radiation, the critical radioactive species is iodine. The steady state levels of iodine in reactor coolant systems are very low, but during an accident they can increase significantly. Current regulatory practice requires that the "spiking" factor during an accident be conservatively estimated at 500. More realistic models suggest that the actual spiking factor is less than this, but additional data and investigation are needed to validate these models.

Assessment of Flow-Induced Vibrations. Although corrosion processes are the primary source of tubing degradation, other degradation processes, such as fretting and wear primarily driven by flow-induced vibrations, also occur in steam generators. EPRI has developed flow-induced vibration codes to aid designers in the analysis of these problems. Further work on this subject is needed to make these codes fully effective.

The high priority projects identified for commencement in FY 1999 are listed below, along with the principal objectives of each project. See Volume II for detailed descriptions of these and other high priority projects in this area. See Volume III for descriptions of all projects in this area.

Project ID:	3-1
Project Title:	Steam Generator NDE Test Mockup Facility
Principal Objective:	To provide suitable test specimens and databases of NDE and characterization results that can be used to develop improved NDE techniques and signal analysis methods for the detection and characterization of SCC in steam generator tubing and provide the capability to assess and demonstrate the effectiveness of such methods.

Project ID:	3-2
Project Title:	Advanced Eddy-Current Inspection System for Detection and Characterization of Defects in Steam Generator Tubes
Principal Objective:	Develop an advanced eddy-current inspection technique and data analysis methodology for more reliable detection and accurate sizing of steam generator tube defects.

Project ID:	3-3
Project Title:	Steam Generator Tubing Diagnostics
Principal Objective:	The purpose of the project research is to improve and automate the assessment of steam generator tubing integrity (correlation of tube burst pressure with inspection signals) through the application of Artificial Intelligence (AI) techniques to Nondestructive Examination (NDE) signal processing. Eddy current (EC) and ultrasonic testing (UT) inspection database information will be used.

Project ID:	3-4
Project Title:	Crevice Concentration and Chemistry Studies
Principal Objective:	To better understand the relationship between bulk feedwater chemistry and crevice chemistry so that corrosion processes can be better characterized and effects of changes in bulk water chemistry explored.

Project ID:	3-6
Project Title:	Development of Advanced Analytical and Imaging Techniques for Eddy Current Inspection
Principal Objective:	The primary objective of this proposed endeavor is to utilize analytical/ computational EM simulations in conjunction with modern data analysis schemes to help improve current state of eddy current NDE for ISI of SG tubing

3.2 Cables

Cable systems, which consist of both cables and connections, provide the path for signals between sensors and the electronics used for the protection and control of a nuclear reactor, for the control and powering of equipment used during normal operation, and in mitigating the effects of accidents. A large amount of cable is used in a nuclear power plant. For example, a typical new BWR contains ~125 miles of power cable, ~600 miles of control cable, and ~10 miles of instrumentation cable. Figure 3-6 shows the transition from a vertical cable tray to a conduit in a typical commercial nuclear power plant, illustrating how large numbers of cables are bundled together.

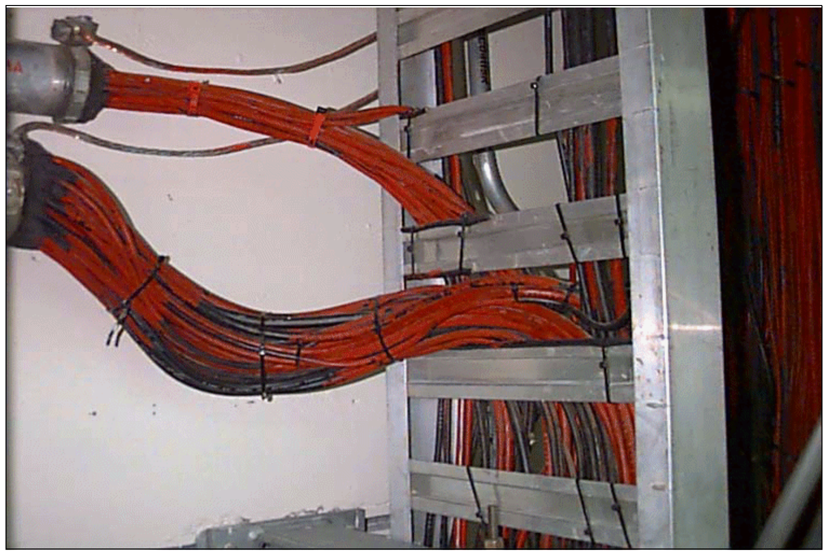

Figure 3-6. Typical Power Plant Cable Installation

Research is needed to establish science-based cable life cycle management tools and techniques; otherwise, plants may have to replace their cable systems, perform additional qualification testing, and/or develop new programs (e.g., condition monitoring). A 1993 estimate for cable replacement was ~$80 million per plant. Additional qualification testing for common cable materials, which can only resolve a subset of the aging issues, is estimated to cost ~$2-5 million. The cost of qualification testing for plant unique cable materials would be several times the cost of qualifying the common materials because there are so many different materials used in U.S. nuclear plants. The aging, condition monitoring, and integrated solutions technology areas described herein are new endeavors to establish the needed tools and techniques.

The deliverables from this technology area are expected to eliminate the need for extensive replacements, eliminate additional qualification testing for all but a few materials, and provide tools to resolve all known cable issues. Given a cable's aging behavior, current condition, and the installation environment, uncertainty will be reduced and residual life may be determined. Fully informed life cycle decisions, which ensure adequate safety and minimize cost, can only be made by integrating the results of this technology area, existing qualification testing, and industry experience. Every effort needs to be made to establish a basis to move beyond specific issues by acknowledging valid concerns and demonstrating how the integrated solution addresses these concerns. For example, the accuracy of the Arrhenius model is not a pivotal concern if it can be demonstrated that a cable will perform its safety function for the duration of a plant's operating license.

Planned replacements of limited scope (e.g., materials in hot spots) are being managed and will continue as dictated by plant conditions. While such replacements result in additional operating and maintenance (O&M) expenses, these costs have been and are expected to remain manageable. No additional research is needed to address this issue.

It should be noted that the condition monitoring project described below is expected to provide new nondestructive examination capabilities that will benefit every industry that uses cables in critical applications. Communications systems, complex process controls, weapons, aircraft, ships, spacecraft, satellites, etc. all need the capability to detect small imperfections that may lead to system failures. Recent studies indicate that an NDE technique capable of detecting small imperfections may be possible.

3.2.1 Current R&D for Cables

Current EPRI research related to aging of nuclear power plant cables and assessment of their condition includes a natural vs. artificial aging project and ionizable gas testing research. The natural vs. artificial aging project is establishing material-specific correlations between in-plant natural aging and accelerated aging practices by evaluating cable samples which have been placed in nuclear power plants. Valuable data on the condition of common cable types is being developed, along with a basis for validation of aging models.

An ionizable gas technique is being developed for performing electrical testing on unshielded low-voltage cables in conduits, a common installation in nuclear plant containments. The nonhazardous, ionizable gas displaces the air in the conduit, which reduces the ground plane and allows enhanced performance of electrical testing on unshielded cables. Although this technique has demonstrated that insulation damage can be detected at reduced test voltages, additional development is required to determine if it can be an effective field technique.

A recently completed report, TR-106108, "Diagnostic Evaluation of Low-Voltage Electrical Cables", provides an overview of existing condition assessment technologies and identifies appropriate tests for preservice and inservice inspections, aging assessment, troubleshooting, and failure assessment of cables.

3.2.2 R&D Needs for Cables

Develop Empirical Data to Characterize Aging Degradation of Polymers Used in Electrical Cable. As plant operating experience accumulates and additional research is performed, aging behaviors that could not be predicted from early experiments and environmental quality (EQ) testing are being identified. Figure 3-7 shows highly irradiated cables (lower right of photo) and new cables (upper left of photo) from one of the Department of Energy Savannah River plants. Although the insulation damage at Savannah River was more severe than expected at a commercial plant, and most currently installed cable materials will perform their safety function for many years, uncertainties related to long-term cable system operability and Arrhenius methodology calculations of qualified life values are a significant regulatory concern. A

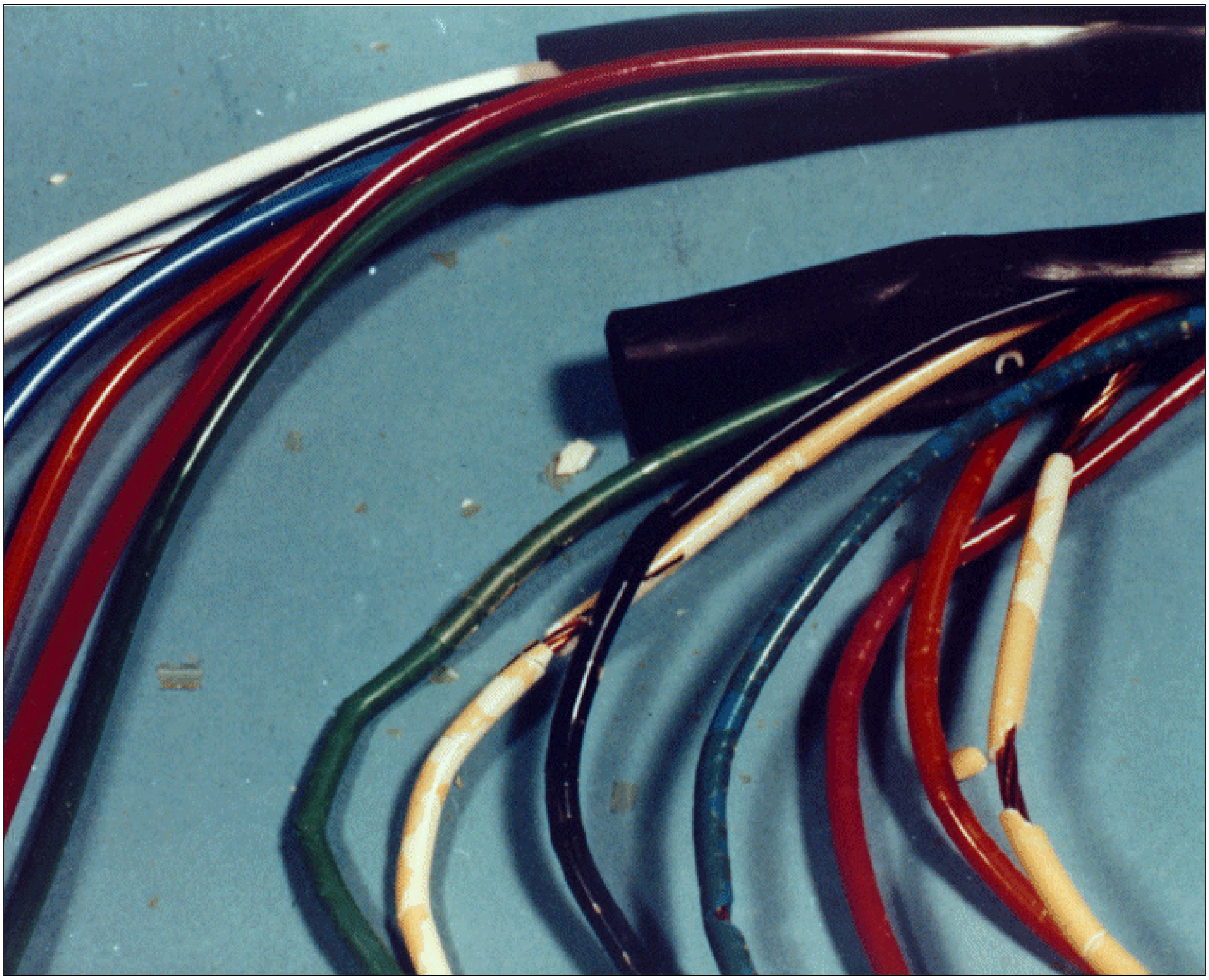

Figure 3-7. Savannah River Low-Density Polyethylene Cables

science-based understanding of polymer degradation is needed to address NRC safety concerns and identify and resolve long-term operational issues.

Compare Natural Aging to Model Predictions Based on Accelerated Aging.

EPRI/UConn Cables - An EPRI/University of Connecticut (UConn) project was initiated in 1985 to establish a material-specific correlation between natural (*in situ*) aging and accelerated aging for four commonly used cable types.[2] Cable bundles, radiation monitors, and temperature monitors were placed in different locations at several power plants. In addition, each participating utility placed plant-specific cable types in its bundles. Naturally-aged cable samples have been removed periodically and sent to UConn where weight, elongation-at-break, and density data are collected

[2] Three of the types manufactured by Okonite, BIW, and Kerite have ethylene-propylene copolymer (EPR) insulation and chlorosulfonated polyethylene (CSPE)/Hypalon jacket, and the fourth type manufactured by Rockbestos has cross-linked polyethylene (XLPE) insulation and Neoprene jacket.

and compared to data from artificially-aged samples. Carried to completion, these tasks will yield a robust data set for confirming and updating the aging models used in existing EQ programs.

Plant Operational Cables - Cable samples from decommissioned plants (e.g., Yankee Rowe), cable replacement activities (e.g., Big Rock Point, before the August 1997 shutdown), and cable monitoring programs (e.g., Oconee) have been gathered. These materials will be evaluated to expand the data set from the EPRI/UConn Cables task. Since these cables were removed from control, instrumentation, and power circuits, operational affects such as power cable self-heating can be studied using these cables. EPRI/UConn cables are not electrically connected; however, they are in the same external environment as plant cables.

Obtain Naturally-Aged Samples. Materials from nuclear power plants and DOE facilities are needed to cover commonly used cable materials. The Low Intensity Cobalt Array (LICA) facility at Sandia (shown in Figure 3-8) can subject cable samples to long term radiation exposure with electrical excitation and appropriate thermal and gas environments. As additional materials and the corresponding environmental pedigree are gathered, the samples need to be input to the naturally-aged cable studies.

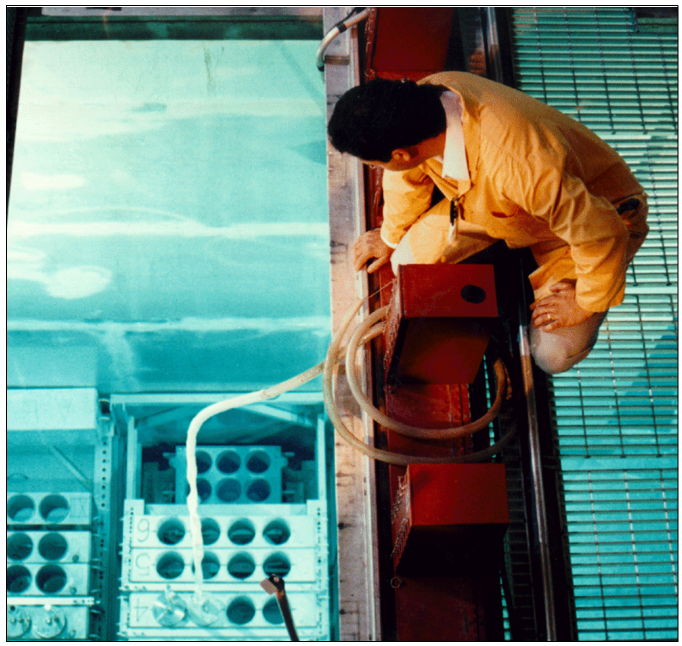

Figure 3-8. Low Intensity Cobalt Array (LICA) For Radiation and Thermal Aging

Confirm/Develop Aging Models. Recent studies provide conclusive evidence that the "combined-environments methodology"/Arrhenius equation will accurately predict natural aging from accelerated aging data for certain combinations of polymers and environments. Studies to date indicate that certain materials have a greater than 60-year life in power plant environments; however, their natural aging behavior is not accurately predicted by the "combined-environments methodology"/Arrhenius equation. Materials exhibiting anomalous aging behavior must be evaluated to ensure that all have adequate (safe) lifetimes. Accurate, material-specific, aging models will provide a sound basis for applying current practices or the foundation for new approaches. Accurate aging models will also be needed to establish meaningful acceptance criteria for condition monitoring examinations and predict remaining life.

Investigate Bonded Jacket Cable Failure Mechanisms. A scoping study performed for NRC revealed a previously unknown failure mechanism associated with bonded jacket cables where cracks in the jacket propagate through the insulation. Additional testing is required to identify whether the scoping study failures are an artifact of the accelerated test regime. Assuming this failure mechanism does not occur in the range from normal operation to design basis event conditions, the existing qualification basis will be confirmed.

While cable jacket integrity is required in special cases, most cable jackets are provided to protect the insulation during installation and the basis for the cable's qualification does not require the jacket to remain intact as the cable ages. However, since the jacket material used in these cables is not as robust as the insulation material, the details of the new failure mechanism must be defined.

Define Similarities/Differences Between Single and Multiple Conductor Cable Aging. A study performed for NRC included one case where a single conductor cable passed the test and the corresponding multiple conductor cable failed the same test. The NRC has stated that additional testing is required for the few cases where multiple conductor cables were qualified by testing a single conductor cable in combination with "type" tests demonstrating the similarity between single and multiple conductor cables.

Develop Condition Monitoring (CM) Techniques for Electrical Cable. Condition monitoring can be used to assess the current condition of a cable and, with evaluations based on empirical data, provide assurance that a cable has sufficient remaining life to perform its safety function(s). Condition monitoring (CM) methods that measure a mechanical, physical, or chemical property at one or more specific locations in a cable run (i.e., sampling techniques) are needed to provide data that quantify the local mechanical condition of insulation and jacket materials. From experience and numerous tests, a threshold mechanical condition has been associated with electrical functional capability; however, sampling tests are best used to identify precursor mechanical conditions (e.g., insulation hardening prior to cracking) that may affect electrical functionality. CM research is needed to:

- Develop a basis, with material-specific correlations between NDE data and destructive examination (e.g., elongation-at-break) data, for localized (sample) inspections
- Develop electrical NDE techniques capable of detecting incipient defects along an entire cable run
- Develop NDE techniques suitable for implementation at nuclear power plants

Cable CM tasks have been grouped into three general categories; existing concepts, new concepts, and data sharing.

Existing Concepts.

Develop a Guideline for Sensory Inspection. Sensory (i.e., visual and tactile) inspection can identify anomalous cable and termination conditions. A comprehensive sensory inspection guide needs to be developed for use by field personnel. The guide needs to define inspection focus and attributes, frequency of inspection, integration of cable inspection into maintenance practices, documentation, and inspector training.

Develop a Comprehensive Data Set for OIT, OITemp, Density, and Indenter. Data collected from other tasks needs to be reviewed to identify any gaps in baseline data for commonly used cable materials. Field use of NDE techniques based on oxidation induction time (OIT), oxidation induction temperature (OITemp), density, and/or indenter needs to be based on a comprehensive collection of test data. As required, material-specific samples need to be artificially aged and tested to cover the required range of environmental conditions. After analyzing the test data, additional inputs need to be incorporated in other cable tasks.

Evaluate Existing Electrical NDE Techniques. Electrical NDE techniques (e.g., DIACS, partial discharge) need to be evaluated at a proof testing facility to determine whether these techniques are capable of detecting incipient defects.

Develop Sample Removal Techniques. Nuclear power plants need microsample removal techniques that do not affect cable qualification while ensuring that representative samples are collected and cataloged. Sampling techniques and tooling need to be developed and qualified.

New Concepts.

Investigate Modulus Profiling. Whether modulus profiling is a suitable basis for a CM technique needs to be determined. Recent studies indicate that modulus profiling may be as effective as the indenter, density, and/or OIT(OITemp). Further studies are needed to investigate common polymer materials.

Develop Electrical NDE Techniques. An electrical NDE technique capable of detecting and locating incipient defects in an entire cable run is needed. A test during normal plant conditions needs to find defects that may fault due to the reduced ground plane that will be present during a design basis event. While efforts to reduce the ground plane (e.g., flooding with water or ionizable gas injection) in combination with existing electrical techniques have successfully located defects in the laboratory, most plant systems cannot be tested in this manner. Research is needed to develop a robust electrical test technique. Test techniques developed under this task also need to be evaluated.

Determine Whether Electrical NDE Techniques Are Nondestructive Cables used to perform CM tasks need to be destructively examined to ensure that they are not damaged by electrical NDE techniques. High potential ("continuous" and pulse), partial discharge, and insulation resistance electrical inspection techniques need to be investigated to determine whether conditions just below the fault threshold may damage the polymer. Research is needed to determine whether electrical NDE techniques degrade the insulation and/or jacket of a cable system.

Develop Models for Electrical Breakdown Phenomenology. At present, there are no theoretical models for predicting electrical breakdown due to defects in polymers. Experts in the field believe it may be possible to develop a combination of analytical tools and test data to characterize the electrical behavior of polymer defects. Research is needed to provide a basis for acceptance criteria used with electrical NDE techniques.

Data Sharing.

CM Database. Cable CM data (e.g., indenter modulus, density, and oxidation induction time) from research organizations, universities, and utilities need to be compiled for use in comparisons to cables in the field or removed from service. CM data needs to be stored in an electronic format designed to facilitate use by plant engineers and researchers. The database will promote consistency in measurement techniques, data collection and interpretation of laboratory and field results.

Environmental Monitoring (EM) of Cable Systems. Radiation and temperature hot spots exist in commercial nuclear power plants (e.g., an insulation panel could come dislodged and surrounding cables would experience an elevated service temperature until the panel is reinstalled). Since the life of a polymer is directly dependent on the operating environment, tools are needed to identify and characterize plant hot spots.

<u>Develop a Localized Environments Guideline</u>. This guideline needs to investigate plant best practices for identifying and mitigating localized high stress environments. For cables and terminations, high stress environments include thermal and radiation hot spots and caustic environments.

Integrated Solutions to Cable Issues. The effort to resolve cable issues includes three base technology projects covering; aging behavior of cable materials, cable condition monitoring (CM), and plant environmental monitoring (EM). While the base technology can be developed independently, effective cable life cycle management relies on the integration of all three projects, plus additional information from screening evaluations to determine the safety significance of a cable's function(s). This project will also provide an opportunity to focus limited licensee and regulatory resources on safety significant applications.

<u>Develop a Cable Life Cycle Management (LCM) Guideline</u>. A consistent approach to screening cable and terminations based on their functional importance and/or vulnerability to failure from aging degradation is needed. Guidance for identifying cable and terminations that should be included in aging evaluations or CM/EM programs needs to be developed. In addition, guidance is needed to define locations, sampling methodology, etc. for CM/EM programs and to establish strategy options for cable life cycle management.

<u>Develop a Cable Life Cycle Management (LCM) Training Course</u>. Training course materials need to be produced for plant personnel. Course materials need to cover cable and termination aging stressors and effects, techniques to mitigate aging stressors, inspection attributes, condition assessment and monitoring, environmental monitoring, documentation, and failure evaluation.

<u>Develop Methods to Predict Residual Life</u>. Techniques to predict the residual life of a cable or termination based on an aging model, knowledge of a component's current condition and operational environment, and margin required at end-of-life to ensure functionality during a design basis event need to be developed.

<u>Transfer CM Basis Technology to Industry</u>. The Condition Monitoring project will produce basis information for CM techniques and lab-scale CM techniques (i.e., the viability of using a parameter like density will be fully supported by experimental data). Technology transfer/licensing agreements will be developed to transfer technical data to companies that implement CM techniques (e.g., the EPRI/Ogden collaboration on the cable indenter). Technical support will be provided for demonstration activities through the field trial stage at power plants.

New technologies (e.g., modulus profiling, electrical pulse techniques) will be licensed and will hopefully provide a revenue stream to support future activities.

Provide Technical Input for Regulatory Updates. Semi-annual meetings with the Nuclear Regulatory Commission (NRC) are needed to ensure that the latest cable aging, CM, and EM information is fully communicated. These forums will also provide a means to ensure that NRC efforts to resolve generic safety issue GSI-168 account for the latest technical developments. The objective of this effort is to support resolution of GSI-168, including license renewal issues.

Provide Technical Input for Codes and Standards Updates. The joint DOE/EPRI effort needs to provide the technical basis for codes and standards updates, including; IEEE-323, IEEE-383, IEEE-1205, and International Electrotechnical Commission (IEC) standards.

Support International Cooperation. The joint DOE/EPRI effort needs to be coordinated with the International Atomic Energy Agency (IAEA) Coordinated Research Programme (CRP) on Managing Aging of Nuclear Power Plant Cables and the Organization for Economic Co-Operation and Development (OECD) Plant Lifetime Improvement (PLIM) Program.

The high priority projects identified for commencement in FY 1999 are listed below, along with the principal objectives of each project. See Volume II for detailed descriptions of these and other high priority projects in this area. See Volume III for descriptions of all projects in this area.

Project ID:	3-6
Project Title:	Develop Empirical Data to Characterize Aging Degradation of Polymers Used in Electrical Cable
Principal Objective:	Develop empirical data to characterize the aging behavior of polymer materials in electrical cable insulation and jackets for the following environments; typical power plant conditions, R&D laboratory experimental conditions, and environmental qualification tests.

Project ID:	3-7
Project Title:	Develop Condition Monitoring (CM) Techniques for Electrical Cable
Principal Objective:	Develop nondestructive or essentially-nondestructive, science-based, CM techniques for electrical cable insulation and jacket materials that are capable of characterizing the current condition of either a local section or an entire cable run using parameters (e.g., density) correlated to aging models or other well-defined criteria.

Project ID:	3-8
Project Title:	Integrated Solutions to Cable Issues
Principal Objective:	The purpose of this Integrated Solutions project is to ensure that complementary empirical aging data, condition monitoring (CM) techniques, and plant environmental monitoring (EM) methods are developed as options to meet the intent of regulatory requirements, address unresolved technical concerns, and support codes & standards updates, if required.

3.3 Reactor Internals

Intergranular stress corrosion cracking (IGSCC) in BWR piping systems, which was a significant problem in the late 1970's, involved "sensitization" of the stainless steel, i.e., depletion of the chromium at the grain boundaries. Neutron irradiation over time can result in susceptibility to IGSCC even in nonsensitized stainless steels because of the redistribution of the impurity and alloying elements at the grain boundaries, and embrittlement of the material. This irradiation induced IGSCC is referred to as irradiation assisted stress corrosion cracking (IASCC). Figure 3.9 shows the increasing predominance of IGSCC on BWR neutron absorber tubes from an operating reactor with increasing neutron fluence.

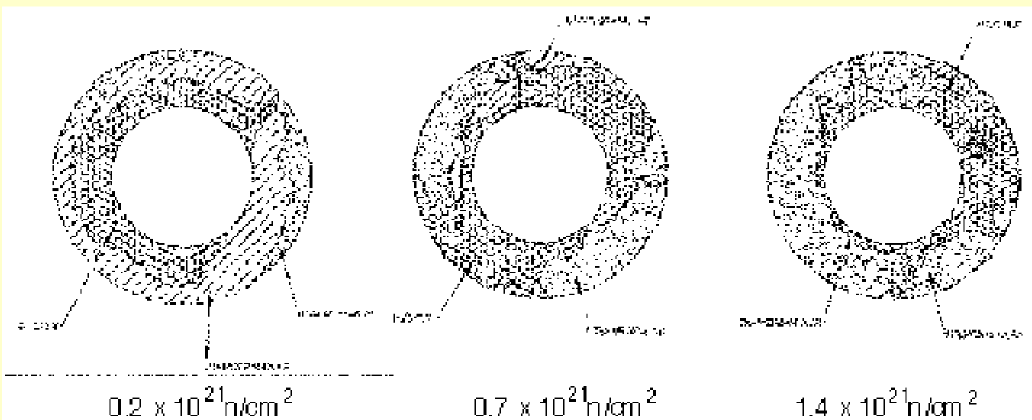

Figure 3-9. Intergranular stress corrosion cracking of BWR neutron absorber tubes as a function of neutron fluence

High radiation levels in a reactor core can also increase the susceptibility of the core structural materials to stress corrosion cracking because of changes in the water chemistry due to the radiolytic decomposition of the water and degradation of the materials themselves. The higher levels of aggressive radiolysis products such as hydrogen peroxide (H_2O_2) increase the electrochemical potential of the coolant. The arrangement of the control rod drive and reactor internals used in the BWR-3 and BWR-4 designs is shown in Figure 3.10. IGSCC, IASCC, and fatigue damage locations are identified in the Figure. Although the PWR water environment is less aggressive than that in BWRs, some PWR core components are exposed to even higher fluences and may also become susceptible to failure, although the actual microstructural mechanisms may be somewhat different than in the case of the BWR.

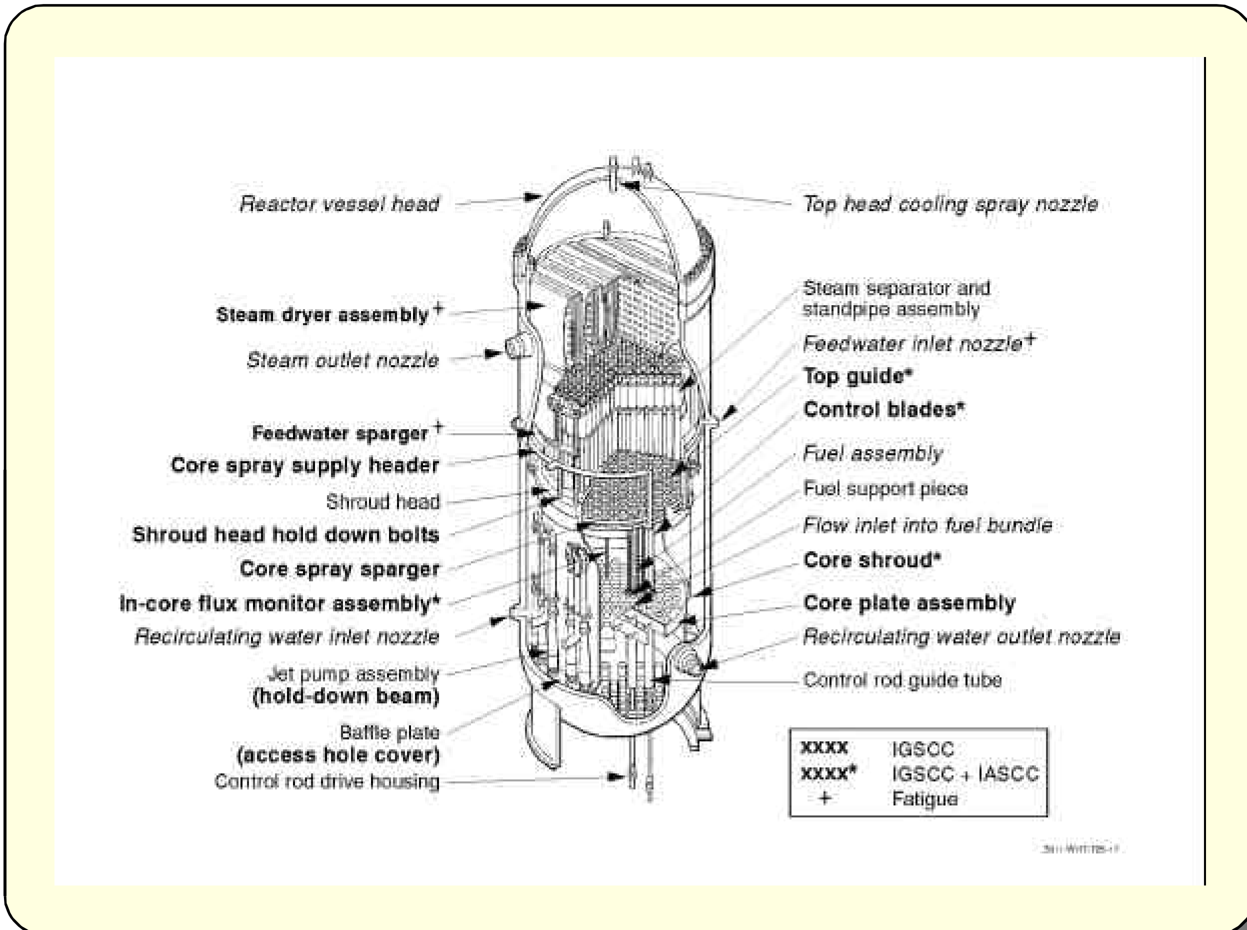

Figure 3-10. Cutaway view of BWR reactor pressure vessel and internals, showing locations of areas susceptible to stress corrosion cracking and fatigue damage

The major issues related to managing plant aging for the reactor internals are: (1) managing the effects of IGSCC for susceptible internals components (e.g., austenitic stainless steel material, a crevice- containing geometry, and sensitized material microstructure) that are exposed to reactor water with sufficiently high electrochemical potential; (2) managing the effects of IASCC for internals components exposed to sufficiently high neutron flux over a sufficiently long time to reduce the material ductility and fracture toughness to levels of concern; and (3) managing the effects of stress relaxation for those internals components (e.g., bolts, pins, and fasteners) that depend upon preload to maintain their function. Other plant aging issues, such as cracking caused by high-cycle fatigue, wear caused by relative motion between parts, and dimensional changes caused by exposure to radiation, are well understood and are of less significance.

The current approach for managing the potential effects of IGSCC, IASCC, and stress relaxation is based on periodic in-service inspection (e.g., visual examination) of internals components that can be removed from the reactor vessel, or internals components that are accessible or can be made accessible for examination, in order to detect any relevant conditions. This approach also includes supplementary examinations to further characterize relevant conditions, plus any corrective actions or engineering evaluations intended to demonstrate fitness for continued service.

3.3.1 Current Research on Reactor Internals

Current R&D on reactor internals is directed at condition assessment (e.g., in-service inspection and engineering evaluation of relevant conditions) of components in operating plants, prevention or mitigation of the service conditions that might lead to degradation, and development of replacement materials that are less susceptible to degradation.

<u>Condition Assessment Guides for BWR Internals</u>. BWR owners have organized a collaborative effort, the BWR Vessel and Internals Project, (BWRVIP) to develop methodologies to manage the degradation of reactor internals. The BWRVIP is developing a comprehensive set of guides that will provide utilities with the information needed to make cost-effective decisions to manage degradation of the BWR reactor vessel and internal components. These condition assessments will cover in-service inspection procedures, repair or /replacement techniques, potential mitigation methods (e.g., water chemistry guidelines), and overall assessment of fitness for continued service.

<u>Engineering Evaluation of IGSCC.</u> R&D on crack growth rates is being performed to support assessments of serviceability and service life when SCC is indicated by inspection, to avoid unnecessary or premature repair/replace decisions and to quantify the benefits of countermeasures such as hydrogen water chemistry (HWC) or noble metal chemical addition (NMCA). Three separate initiatives have been taken to collect and correlate data on stainless steels, on nickel-base alloys, and on low alloy steels, respectively. The NRC is also supporting work to determine IASCC crack growth rates.

Stress corrosion is influenced by residual stresses, which are typically high at weldments. BWRVIP has supported residual stress measurements and analyses on typical welded construction of key BWR internals. BWRVIP also supports development of analytical models predicting the effects of gamma and neutron fields on chemical dissociation and recombination in BWR coolant circuits. And, the BWRVIP has continued prior EPRI developments on processes and consumables for repair welding, including underwater welding.

<u>Mitigation Methodologies</u>. Since the early 1980s, BWR plants have had the option of adding hydrogen to the reactor coolant to suppress stress corrosion cracking. The noble metal chemical addition (NMCA) process developed by General Electric produces a very thin durable deposit of platinum and rhodium on reactor internal components that improves the effectiveness of hydrogen water chemistry by enhancing the recombination of radiolytic oxygen with injected hydrogen. The application of NMCA was demonstrated by the BWR VIP in the Duane Arnold BWR in October 1966. Ongoing work includes monitoring of electrochemical potential and crack growth over two cycles and fuel surveillance over three cycles. The NMCA fuel surveillance plan also includes post-irradiation examination of discharged fuel in a hot cell to look for evidence of increased hydrogen uptake in the fuel cladding which could be a precursor to hydriding or hydrogen embrittlement. Water-jet peening to mitigate IGSCC in BWR and PWSCC in PWR plants is being investigated under a DOE small business innovative research contract.

<u>IASCC Research</u>. The Cooperative IASCC Research program (CIR) is a five-year program managed by EPRI to develop a mechanistic understanding of IASCC, to develop a predictive methodology based on this understanding, and to identify possible countermeasures to IASCC.

Collaboration is motivated by the high cost of research on irradiated materials. International sponsors of the CIR program include the NRC, other foreign regulatory authorities, NSSS vendors, nuclear utilities and utility organizations. A current focus is collection and analysis of existing laboratory and field data related to IASCC. A database has been developed and is periodically updated. An in-depth literature study has been published. Key technical issues have been identified to guide the research. Work is in progress on a systematic in-depth study of irradiated materials known to be either susceptible or resistant to IASCC. Materials are being tested in both aggressive and benign environments found in both BWRs and PWRs. Critical testing parameters and acceptance criteria will be established for qualification of replacement materials.

Stress Corrosion Cracking in PWR Alloy 600 Penetrations. The Nuclear Energy Institute (NEI) has organized an ad hoc task group to resolve issues associated with stress corrosion cracking in PWR Alloy 600 vessel head penetrations. EPRI has a multi year research project that began in 1993 to provide data and assessment methods to predict the residual life of Alloy 600 penetrations in PWRs, including crack growth rates as a function of microstructure, stress and water chemistry, and the effect of microstructure on crack initiation in penetration materials and evaluate remedies such as shot peening and zinc additions to the PWR primary water.

Procedures and hardware for laser weld repair of PWR vessel head penetrations damaged by stress corrosion cracking are also being developed.

Replacement Material for Baffle/Former Bolts. PWR core internals include structures called baffles and formers that surround the reactor core (shown in Figure 3-11). The baffle/former bolts that hold these structures together are exposed to high neutron radiation levels, and have failed in a number of plants in the U.S. and overseas. Research related to this emerging issue is focused on characterization and prediction of the service performance of existing and replacement materials. Candidate materials will be screened in a power reactor and performance and life prediction data will also be generated.

3.3.2 R&D Needs for Reactor Internals

Fundamental research on the characterization of changes in material behavior of austenitic stainless steel caused by high levels of irradiation is warranted, to improve the basis for in-service examination, engineering evaluation, mitigation, and repair/replacement of reactor pressure vessel internals. This additional research will involve detailed examination of exposed materials removed from service, along with micro-structural material studies and development of models (e.g., IASCC crack growth models) with features beyond those being considered by the industry in current research programs. Some of this fundamental materials research will also apply to related issues, such as IGSCC and stress relaxation. The broad outline of this additional research is described in the following paragraphs; more information is provided in the detailed task descriptions in the Appendix.

Fundamental Characterization of IASCC. Through the EPRI BWRVIP program, industry has developed an empirical model that can predict crack growth rates in unirradiated materials as a function of loading and water chemistry conditions. However, irradiation produces significant changes in the microstructure and mechanical behavior of stainless steels and that model is probably only applicable to components such as the core shroud that are exposed to relatively low fluences. Although additional empirical data are needed, testing of irradiated materials is difficult and expensive, and development of an empirically based model for materials subjected to high fluences akin to those developed for unirradiated and low fluence materials is impractical. An improved, physically based understanding of degradation processes and their effects on crack initiation, crack growth rates, stress relaxation, and fracture toughness is necessary.

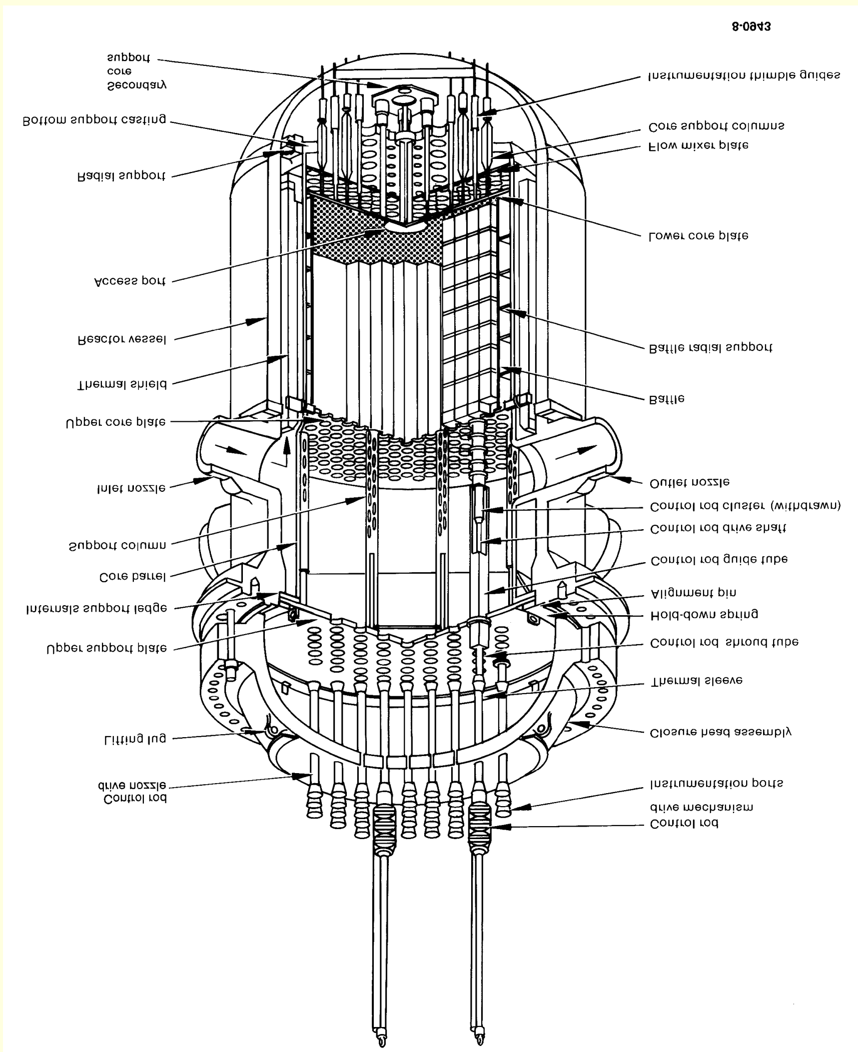

Figure 3-11. PWR core internals, including baffles and formers

This understanding is needed for materials exposed to a whole range of neutron fluences ranging from relatively low values (core shrouds in BWRs), to moderately high values (top guides in BWRs), and very high values (baffle bolts in PWRs). There is an ongoing international cooperative effort in this area managed by EPRI and studies proposed here will be coordinated with that work.

Less-Susceptible Materials. Stainless steels show a wide variation in resistance to IASCC. However, it is difficult at present to specify the composition and fabrication conditions that will assure that a material of interest is one of the more resistant materials. EPRI sponsored work has led to the development of a material specification for improved Type 348 stainless steel for thin-section, in-core applications, but at present it is not possible to specify the composition and fabrication conditions that will ensure high resistance for components that are not fabricated from thin sheet materials. Work is needed to identify more resistant materials for such components.

IASCC Weld Repair Methods. When continued structural integrity cannot be ensured, decisions will have to be made about repairing or replacing components. Replacing components is generally an option of last resort due to the high radiation dose rates and costs involved in such an operation. In situ repairs to date have relied on mechanical repairs (clamps, etc.), whose applicability is limited to certain component geometries, because welding of highly irradiated stainless steels is very difficult. There has been substantial work by the industry to develop techniques for underwater welding of stainless steels, leading to the development of industry consensus standards. However, to date such techniques have been successfully applied only for components with relatively low fluence.

During neutron irradiation of stainless steels, helium is formed by neutron reactions with some of the alloy constituents. Helium has a low solubility in metals, and tends to form small clusters or bubbles. At elevated temperatures, these clusters and bubbles can grow and migrate to grain boundaries very rapidly and severely weaken these boundaries. Because conventional welding processes involve high temperatures and result in significant stresses, irradiation-induced helium can severely affect the weldability and the postweld properties. The DOE fusion reactor research program has been actively investigating this problem and substantial progress has been made in developing a thorough understanding of the physical processes that cause such cracking. This has led to some promising techniques for solving the problems encountered while welding high fluence materials.

However, in the fusion-related research the focus has been on welding in dry environments and hence the work is not directly applicable to the welding problems of interest for light water reactors. Also, work has been done at Savannah River on this problem. Additional research is needed to develop an underwater welding technique for irradiated stainless steels that is amenable to remote execution, can be used for an actual load-carrying structural repair (rather than simply a superficial covering of a crack), provides adequate inspectability of the repair, and will have ASME code recognition and qualification. Such work will draw on industry experience in remote underwater welding techniques, DOE national laboratory experience in developing welding techniques for irradiated materials in the fusion reactor and defense programs, and DOE national laboratory capabilities to handle highly irradiated materials. Because of the expense and difficulty of working with highly irradiated materials, both experimental studies and analytical studies of the welding of these materials may be pursued.

<u>IASCC/IGSCC Mitigation Methods.</u> High temperature surface annealing of BWR vessel internals surfaces using a plasma heat source can dissolve chromium carbides and eliminate stainless steel sensitization. Associated fast cooling from the annealing temperature will avoid formation of new carbides on the surface. Laboratory results show that such an annealing process can eliminate surface sensitization completely. Further work is needed to develop field portable equipment and techniques.

The high priority projects identified for commencement in FY 1999 are listed below, along with the principal objectives of each project. See Volume II for detailed descriptions of these and other high priority projects in this area. See Volume III for descriptions of all projects in this area.

Project ID:	3-12
Project Title:	Development of Physically Based Models for Irradiation Assisted Stress Corrosion Cracking (IASCC)
Principal Objective:	Develop physically based models of IASCC that can be used to help understand the effect of critical variables such as alloy composition, fluence, stress level, and grain boundary structure on susceptibility to IASCC and crack growth rates.

Project ID:	3-13
Project Title:	Mechanical Behavior of Irradiated Structural Stainless Steels
Principal Objective:	The objective of the work is to determine the mechanical behavior of irradiated structural stainless steels under conditions of interest to LWRs and to develop constitutive models describing that behavior that can be used in conjunction with research on irradiation-assisted stress corrosion cracking (IASCC), intergranular stress corrosion cracking (IGSCC), helium embrittlement, and welding of irradiated materials to develop tools to predict component life, assess the results of NDE examinations and guide the timing of corrective actions.

3.4 Reactor Pressure Vessels

The reactor pressure vessel (RPV) is arguably the most important single safety-related component in a nuclear power plant. Demonstration of RPV integrity (determined principally by its resistance to brittle fracture) is critical to ensure continued, safe plant operation. A typical PWR RPV is shown in Figure 3-12 with the important subcomponents identified. The major issues related to managing plant aging for the RPV are: (1) managing the effects of neutron radiation induced embrittlement for those portions of the vessel (e.g., the beltline region) that are exposed to sufficiently high neutron flux over a sufficiently long time to reduce the fracture toughness of the beltline materials to levels of concern; and (2) managing the effects of primary water stress corrosion cracking (PWSCC) for vessel penetrations, such as those for insertion of instrumentation and control rods. Other plant aging issues associated with the vessel, such as cracking of the vessel interior cladding, fatigue cracking of nozzles, and stress corrosion cracking of closure studs, are of less significance.

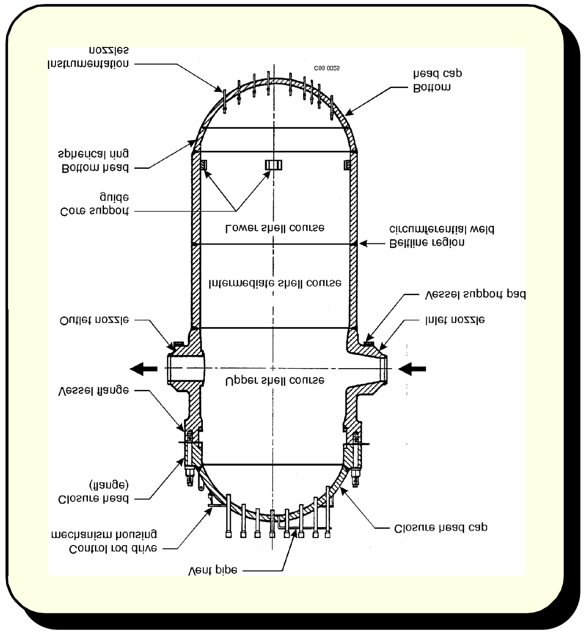

Figure 3-12. Cross section of typical Westinghouse reactor pressure vessel

The current approach for managing the potential effects of neutron irradiation embrittlement is based on a combination of: (1) periodic inservice inspection (e.g., volumetric examination by ultrasonic methods), in order to detect and size any significant pre-service or service-induced flaws; (2) beltline material surveillance, using duplicate material specimens exposed to an accelerated neutron irradiation environment (i.e., lead specimens), to detect significant reductions in fracture toughness; and (3) potential operational controls, as required by regulations. The combination of these three program elements are used to evaluate the fitness of the RPV for continued service. Therefore, the issue of managing the effects of neutron irradiation embrittlement encompasses the sub-issues of the adequacy of the material surveillance program, the satisfaction of pressurized thermal shock regulations, potential low upper shelf toughness, and potential pressure-temperature operational limits.

The current approach for managing the effects of PWSCC on RPV penetrations primarily involves periodic in-service inspection (e.g., visual examination) to detect leakage from through-wall cracking in Alloy 600 material, plus augmented inspection (e.g., surface or volumetric examination) and other corrective actions to characterize and remediate relevant conditions of concern.

The current approaches to RPV integrity assessment (including NRC regulations and methodologies of standards organizations) were originally implemented incorporating a high degree of conservatism to account for uncertainties. As plants age, operation of the RPV may become unnecessarily restricted due to the conservative nature of existing regulations and assessment procedures. By utilizing advancements in material characterization and integrity assessment technologies, a more realistic determination of RPV integrity can provide the flexibility required by licensees and still maintain adequate levels of safety against vessel failure. As radiation-induced embrittlement of the RPV increases with plant operation, the continued demonstration of RPV integrity will become a critical factor in the decision whether to pursue additional operation under license renewal provisions. Resolution of RPV integrity issues can result in substantial cost savings through reduced regulatory scrutiny, increased operating flexibility, and extended operation.

3.4.1 Current Reactor Pressure Vessel R&D

Current RPV R&D is directed at reducing the uncertainties in the evaluation of the fitness for continued service. The sound, technical demonstration of these improved evaluation techniques will be a factor in ensuring continued operation.

Reactor Pressure Vessel Embrittlement Management. EPRI developed and maintains databases of information on vessel embrittlement. These serve as a repository for industry data and are an aid to identification of inconsistencies in materials. In addition, EPRI developed guidelines to assist utilities in evaluating uncertainties in material properties. EPRI also maintains an embrittlement management handbook to incorporate new EPRI-developed technology and products. EPRI is developing a fracture toughness handbook, including a comprehensive fracture toughness property reference and application primer, to be completed in 1999.

Industry and the NRC are currently supporting efforts to demonstrate and establish the use of the so-called "Master Curve" approach to the determination of fracture toughness reference temperatures. This approach is nearing approval by the American Society for Testing Materials. This approach will provide a more realistic assessment of initial RPV fracture toughness and should reduce unnecessary conservatisms associated with RPV integrity assessments.

To conserve surveillance material, EPRI is developing an alternative approach to the determination of deformation and fracture characteristics referred to as small punch (SP) testing. The SP test uses a small "shirt button" disk of material. EPRI has successfully demonstrated its applicability on turbine rotor materials. A program is presently underway to demonstrate the applicability of the SP technique to testing both unirradiated and irradiated RPV materials for application to RPV integrity assessment. At present the SP technique is not universally accepted for use in RPV integrity assessments. EPRI plans to continue efforts to demonstrate applicability of the SP technique to the integrity assessment of RPVs and to gain regulatory and code-body acceptance of the approach.

EPRI is currently cooperating with NRC on the NDE of sections of a decommissioned RPV in order to develop more representative generic or postulated characterization of flaw density and distribution. Methodologies to predict flaw distributions based on vessel fabrication techniques and procedures are also being developed. EPRI will pursue a strategy to combine these technologies to develop more realistic estimates of flaw distributions.

RPV Pressure and Temperature Limit Optimization. To ensure an adequate margin against fracture, limits are placed on the internal pressure during heating and cooling of the vessel. The allowable pressure-temperature (P-T) limits for reactor heatup and cooldown are defined by Appendix G to Section XI of the ASME Code for the beltline region of the reactor pressure vessel. Recent ASME XI activities based on EPRI and NRC research have focused on developing revisions to Appendix G to allow the use of improved analysis methods with more explicit definition of the assumptions and the intended margins that are to be included in the analysis. EPRI-supported work includes the development of a risk-informed flaw-tolerance methodology to provide a technical basis for relaxing current Appendix G flaw-size assumptions. This will expand available P-T operating windows and reduce current reactor heatup and cooldown rate restrictions.

Reactor Vessel Thermal Annealing Methods. EPRI is supporting RPV annealing demonstrations to provide utilities with an option for significantly extending the operating lifetime of PWR plants. The principal issues for annealing U.S. plants are (1) the level of dimensional distortion that could result from the process, (2) the capability to predict vessel thermal response, and (3) the reembrittlement rate of materials following the anneal. Development of an annealing recovery and reembrittlement database has also been initiated to optimize material performance following an anneal.

Irradiation-Induced Changes in RPV steels. In collaboration with CRIEPI (a Japanese nuclear energy consortium), EPRI is supporting fundamental studies of radiation embrittlement of RPV steels using state-of-the-art microstructural techniques to provide insight into the mechanisms that cause radiation embrittlement. NRC is also sponsoring similar fundamental studies at the University of California, Santa Barbara and at the Oak Ridge National Laboratory. An understanding of these mechanisms as reflected through changes in the steel microstructure can be a valuable tool in developing predictive micro-mechanical models. It is expected that this work will lead to improved physically based trend formulas for predicting radiation embrittlement. This research project will continue through 1999, resulting in a series of reports on irradiation-induced changes and embrittlement behavior of RPV steels.

Resolution of PTS Issues. An analysis of the Yankee Rowe RPV highlighted the shortcomings of present methodologies for evaluating pressurized thermal shock as outlined in NRC Regulatory Guide (RG) 1.154. The uncertainties associated with the probabilistic fracture mechanics (PFM) and thermal-hydraulics analyses resulted in very conservative estimates of vessel failure probabilities that would have required extensive plant modifications. In response to this issue, EPRI recently developed an alternative approach to evaluation of PTS, which is intended to more effectively evaluate the impact of corrective measures such as flux reduction, plant modifications, and changes in plant operating procedures on susceptibility to PTS. A plant demonstration of the methodology is under way, and results will be used to develop a proposed revision to Reg. Guide 1.154 in 1999. EPRI will continue to work with others in the industry to gain acceptance of the methodology by the NRC for incorporation into a future revision of RG 1.154.

3.4.2 Reactor Pressure Vessel R&D Needs

Further research is warranted to reduce the conservatism required in the management of neutron irradiation embrittlement effects and the assessment of the fitness of reactor pressure vessels for continued service. Much of this additional research will rely on fundamental material studies and development of techniques well beyond those under current consideration by the industry. The broad outline of this additional research is described in the following paragraphs, and more information is provided in the detailed project descriptions in Volume II.

There are several potential approaches to reducing the conservatism in the evaluation of the fitness of the RPV for continued service. One is to reduce the conservatisms in fracture mechanics analyses of vessels during operational transients such as plant heatups and cooldowns or during accidents such as pressurized thermal shock. Another is to better characterize the embrittlement in the material and its effect on the fracture toughness of the RPV. A better characterization can be achieved by a better understanding of the fundamental mechanisms of embrittlement, by better testing techniques that can make more efficient use of the limited amount of surveillance materials that are available, and, ideally, by nondestructive examination techniques that will permit the characterization of the actual vessel material of interest rather than empirical correlations based on a generic database of irradiated materials. This is especially important in terms of license renewal because, in many cases, the current surveillance programs may test most of the available specimens during the initial license period and there may not be sufficient specimens available during the period of license renewal if only conventional testing procedures are used.

<u>Reactor Vessel Material Surveillance Program.</u> To optimize the use of available surveillance materials, a variety of small specimen testing techniques or specimen reconstitution techniques have been considered. The EPRI small punch test was discussed in the previous section. Research is proposed on alternative approaches utilizing indentation hardness and ultra small tensile specimens. As shown in Figure 3-13, the ultra small tensile specimen is not much larger than the bow tie on the Lincoln penny. Such small specimens make very effective use of limited quantities of surveillance materials, and greatly reduce the problems of handling radioactive materials. The indentation approach has the advantage that it conceivably could be applied in the reactor in an essentially nondestructive manner to obtain the properties of the actual materials of interest even if surveillance materials are not available. The ultra small tensile specimen approach utilizes not just a characteristic property, but the more complete deformation behavior, and could provide an improved correlation with the property of most interest, the fracture toughness.

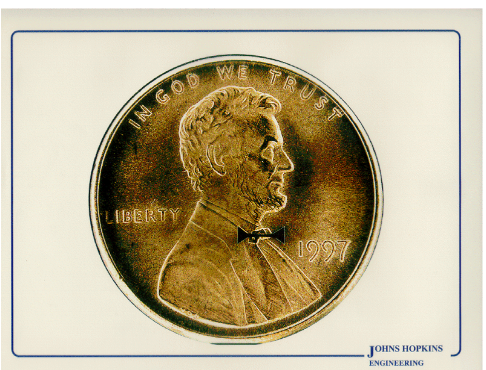

Figure 3-13. Ultra small tensile test specimen

State-of-the-art microstructural techniques can be used to provide valuable insight into the mechanisms that cause radiation embrittlement. An understanding of these mechanisms as portrayed through changes in the steel microstructure can be a valuable tool in developing predictive micro-mechanical models. A new tool has recently become available, the DOE 7 GeV

Advanced Photon Source. This machine provides X ray intensities and energies several orders of magnitude higher than previous sources, which makes it possible, for example, to select out and characterize via X-ray scattering only copper precipitates in material containing a variety of other small-angle scatterers such as carbides and defect clusters.

<u>Reactor Vessel In-service Inspection Program.</u> Nondestructive examination techniques would provide the ideal tool to characterize the embrittlement of the RPV. They could be applied to the actual vessel, rather than relying on properties developed by bounding a generic database. The resolution of conventional ultrasonic NDE techniques is on the order of the grain size, about 10 to 2000 microns. Resolution in the submicron range is needed to detect the level of microstructural changes that is responsible for embrittlement, such as precipitates and other defect structures. Initially these changes occur on a scale (20-200 nm) small with respect to the dominant scatterers (typically grains) in the material. For this reason, traditional acoustic NDE techniques, described by linearized scattering theory, do not work well. However, other approaches based on the examination of other ultrasonic properties (attenuation) or by examining the nonlinear response of materials to ultrasonic pulsing could detect defect structures on this size scale and have the potential to provide a nondestructive characterization of the embrittlement.

<u>RPV Pressure & Temperature (P-T) Limit Optimization</u>. The fracture toughness of reactor pressure vessels (RPVs) increases with temperature until an upper shelf region is reached. To ensure an adequate margin against fracture, limits are placed on the internal pressure as a function of temperature during heating and cooling of the vessel. The P-T curve, which defines the upper bound of the P-T operating envelope, has been determined using conservative fracture-mechanics-based criteria and analysis methods. Additional research is proposed to develop more realistic P-T curves using more refined analysis techniques and an improved understanding of the behavior of pressure vessel materials under such conditions. The proposed work complements an ongoing EPRI effort to develop software to simplify the calculation of P-T limits, but would permit additional reductions in the conservatism of the current approach.

See Volume III for descriptions of projects in this area. See Volume II for more detailed descriptions of the high priority projects.

3.5 Structures

Maintaining the integrity of reactor containment structures is required for plant and public safety. Schematics of a large dry PWR prestressed concrete containment and a BWR Mark I type metal containment (enclosed in a reactor building) are shown in Figures 3-14 and 3-15, respectively. The in-service performance of reinforced and post-tensioned concrete structures and metal containments in U.S. nuclear power plants has generally been good. However, incidences of age-related degradation have been reported and may increase due to aging of the materials. Comprehensive in-service performance reviews indicate that damage to the concrete structures is primarily in the form of concrete cracking, corrosion of embedded steel reinforcement and liner materials, larger than anticipated loss of pre-stressing forces, and leaching of calcium hydroxide from the concrete. In addition, post-tensioning tendon wires have failed and tendon corrosion inhibitor has appeared on the outside surfaces of several post-tensioned concrete containments.

Furthermore, due to lack of direct means of monitoring the integrity of containment basemats, the condition of the concrete in these components cannot be easily established. Some modest corrosion of the metal containments, primarily at the wall-basemat interface and adjoining wall locations, has also been observed.

Effective aging-management programs to demonstrate long-term integrity and reliability of safety-related concrete and metal containment structures requires technologies for detecting and quantifying the extent of the damage and an understanding of the degradation mechanisms. Once a problem is identified, remedial actions can be taken to restore a structure's integrity, enhance its reliability, and extend its service life. For this approach to be effective, developmental efforts need to address the enhancement of concrete technologies in selected areas, as well as improvements in metal corrosion detection capabilities. Computerization of a comprehensive materials property database on long-term concrete and related material performance will provide a readily-accessible knowledge base for use in performing durability assessments and service life estimations. Particular emphasis should be focused on the development of improved and validated service life models and acceptance criteria for service life estimates. A damage model is needed that reflects interactions between degradation mechanisms. Such a model will be useful in performing time-dependent reliability analyses in which the failure probability of a degraded component is determined, either at present or at some point in the future. Validation of these analytical tools is considered essential.

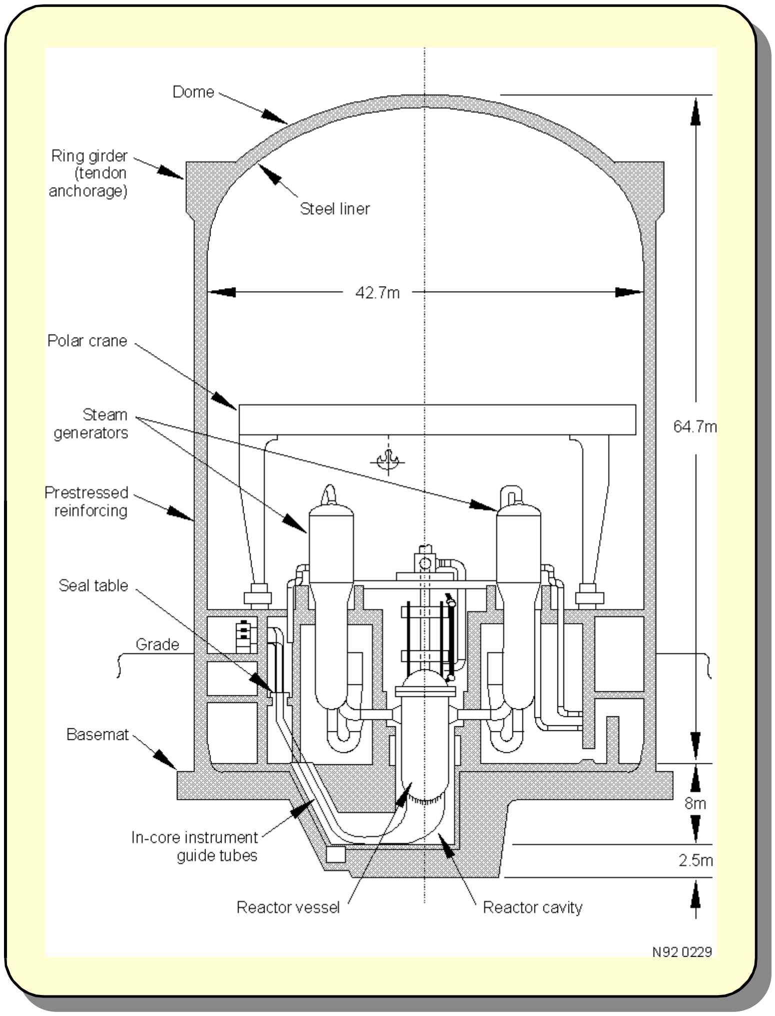

Figure 3-14. PWR Prestressed Concrete Containment

User-oriented guidelines need to be developed so that this technology can be reliably used to determine the effect of aging on structural safety margins. Although repair of reinforced concrete structures is performed routinely, supplementary guidance for the repair of nuclear power plant concrete structures should be developed to address repair strategies and the effectiveness and durability of candidate remedial measures. Currently, there are no NDE techniques capable of providing reliable information about the condition of massive, heavily-reinforced concrete structures such as containment basemats. Techniques capable of inspecting inaccessible regions of the containment pressure boundary also need to be developed and demonstrated. Examination of limited in-service inspection data indicates that the use of lift-off loads to estimate pre-stressing forces in post-tensioned concrete containments may overestimate the actual pre-stressing forces. This concern should be investigated in more detail to determine if this is a significant problem, and if so, to determine the root cause of the problem, its potential impacts on structural margins, and how a more representative estimate of pre-stressing forces can be obtained. The significance of leakage of corrosion inhibitor through cracks in containments should be thoroughly investigated to evaluate its potential effects on the properties of the concrete and overall structural performance.

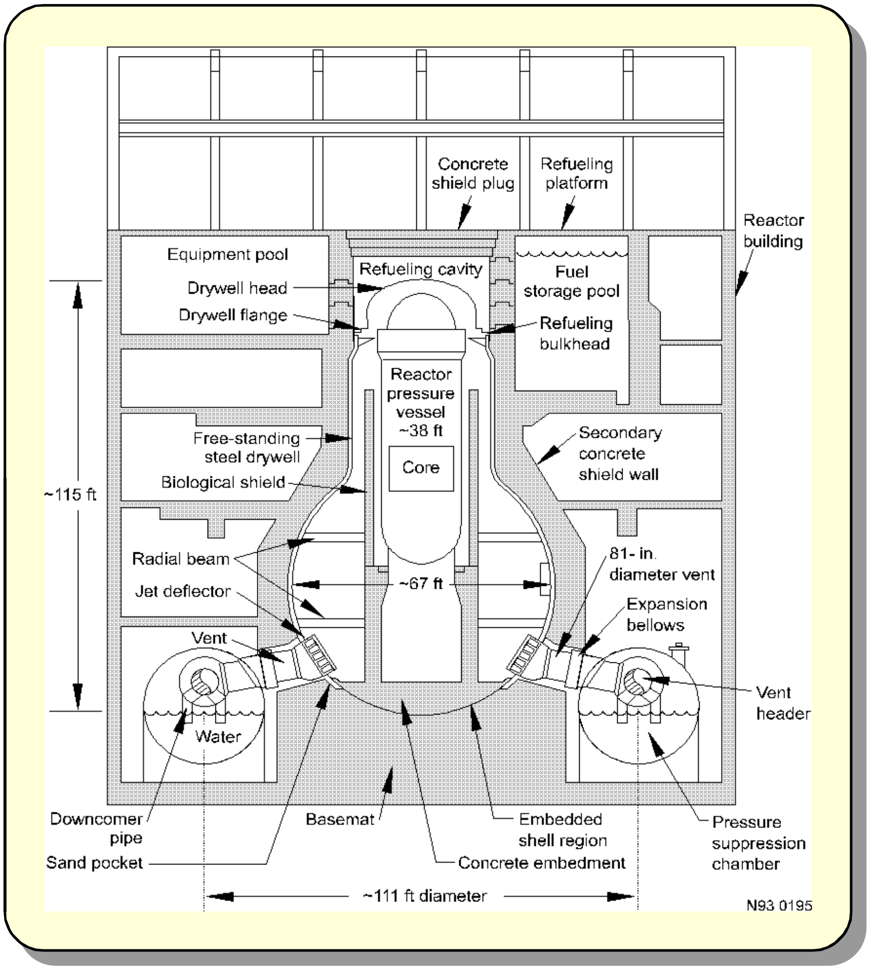

Figure 3-15. BWR Mark I Containment (Inverted Light Bulb and Torus)

3.5.1 Current Research and Development

Issues related to the aging of concrete components and its significance relative to useful life of nuclear power plants were initially identified in the 1980s in reports (EPRI NP-2418, EPRI NP-4208, and NUREG/CR-4652) that were prepared for EPRI and NRC. Based on findings and recommendations presented in these reports, a comprehensive review of issues related to the aging of reinforced concrete structures in nuclear power plants was sponsored by NRC. The overall objective of the Structural Aging (SAG) program was to provide NRC with 1) identification and evaluation of the structural degradation processes; 2) issues to be addressed under nuclear power plant continued service reviews, as well as criteria, and their bases for resolution of these issues; 3) identification and evaluation of relevant in-service inspection and structural assessment programs in use, or needed; and 4) quantitative methodologies for assessing current, and estimating future, structural safety margins. Results of the SAG review were published in a March 1996 report (NUREG/CR-6424) on aging of nuclear power plant concrete structures. Industry reports on Class I structures and PWR containments were also prepared in the early 1990s and published by EPRI. Those reports (EPRI TR-103835 and EPRI TR-103842) addressed license renewal issues and included discussions about effects of concrete aging on long-term performance and safety. EPRI is currently developing an aging management reference manual for condition assessment of concrete structures, and is developing guidelines to assist utilities in satisfying ASME Section XI, IWE/INL inspection requirements. New projects to address the recommendations provided in the SAG reports for additional R&D in the area of aging management of degraded concrete structures in nuclear power plants have not been initiated.

A companion study to the SAG program is currently being conducted at the Oak Ridge National Laboratory. Under this NRC-sponsored program, issues related to the inspection of aged and damaged metal containments and liners of concrete containments are being identified to 1) understand significant factors relating occurrence of corrosion, efficacy of inspection, and structural capacity reduction of metal containments and liners of concrete containments and make recommendations on use of risk models in regulatory decisions; 2) provide NRC reviewers a means of establishing current structural capacity margins or estimating future residual structural capacity margins for metal containments and liners of concrete containments, as limited by integrity; and 3) provide recommendations, as appropriate, on information to be requested of licensees for guidance that could be used by NRC in assessing the seriousness of reported incidences of containment pressure boundary degradation. A complementary research effort is being conducted at Sandia National Laboratories under NRC sponsorship to investigate and develop analytical methods to determine the effects of corrosion on the ability of metal containments to withstand severe accident conditions. Both of these programs are nearing completion.

The effects of tendon corrosion inhibitor (grease) on the tensile and compressive strengths of concrete used to construct post-tensioned concrete containments are being investigated by NRC. This study is focusing on the post-tensioned concrete containment at the retired Trojan nuclear power plant. Concrete cores from the containment have been removed and are being tested to characterize and quantify the degradation, if any, caused by leakage of corrosion inhibitor through cracks in the concrete. Technical information, degradation mechanisms, and field experience related to the management of aging damage to light water reactor metal containments were presented in an NRC contractor report (NUREG/CR-5314, vol. 5) issued in 1994.

3.5.2 Research and Development Needs

Research and development required to understand the effects of aging on the future performance of concrete and metal containment structures in nuclear power plants needs to focus on key areas covering material properties and performance modeling; structural component inspection, assessment, and remediation; and time-dependent reliability service life estimations. These efforts should be integrated into a comprehensive research program that is divided into tasks that address the following topics: 1) development of improved and validated service life models and acceptance criteria for service life estimates; 2) preparation of supplementary guidance for the inspection, assessment, and repair of nuclear power plant concrete structures; 3) development and demonstration of NDE techniques for inspecting concrete and metal components and inaccessible regions of the containment pressure boundary; and 4) development and evaluation of time-dependent reliability service life estimates.

Selected research and development activities should be initiated to establish the models, methodologies, and tools necessary for effective aging management of concrete structures in nuclear power plants. Development of methodologies and models that assess the current condition of degraded concrete components, estimate the extent of damage caused by corrosion, freeze-thaw exposure, and alkali-silica reactions, and predict remaining life is important. For aging-management programs to be effective in demonstrating long-term integrity and reliability of safety-related concrete structures, technologies for detecting and quantifying the extent of damage that has occurred, and for understanding the effects of degradation mechanisms on structural response, should also be developed. Candidate NDE techniques that could be evaluated for use in establishing the condition of degraded concrete components and inaccessible portions of containment pressure boundary components include microwave sensors, ultrasonic techniques, guided plate waves, magnetostrictive sensors, aperture imaging, and electromagnetic acoustic transducers. In order to optimize in-service inspection and maintenance strategies based on risk parameters, a time-dependent reliability methodology should be developed. Once developed and validated, such a methodology would be useful in establishing reliability-based future condition assessments.

The high priority projects identified for commencement in FY 1999 are listed below, along with the principal objectives of each project. See Volume II for detailed descriptions of these and other high priority projects in this area. See Volume III for descriptions of all projects in this area.

Project ID:	3-21
Project Title:	Structural Component Inspection, Assessment, and Remediation
Principal Objective:	Guidelines for inspecting, assessing, and repairing aged concrete structures in nuclear power plants will be developed for use in demonstrating that these structures are safety and reliable for continued service.

Project ID:	3-24
Project Title:	A Guided Ultrasonic Wave Technique for Inspection of Embedded Portions of Metal Containments
Principal Objective:	Develop a technique which uses guided ultrasonic wave modes (Long Range Lamb Waves) for locating, detecting and quantifying corrosion and pitting in the embedded portions of the PWR and BWR metal containment structures.

3.6 Piping Systems

This technology area focuses on methods for managing service-related degradation in piping, pumps, and valves. A schematic of a Westinghouse PWR reactor coolant system, with locations of relatively high fatigue usage identified, is shown in Figure 3-16. The tasks in this area include guidance on risk-based inspection of piping systems, utilizing information on degradation potential, and advanced methods for assessing and monitoring fatigue and thermal aging damage. Performance and reliability of balance-of-plant (BOP) heat exchangers are important current issues.

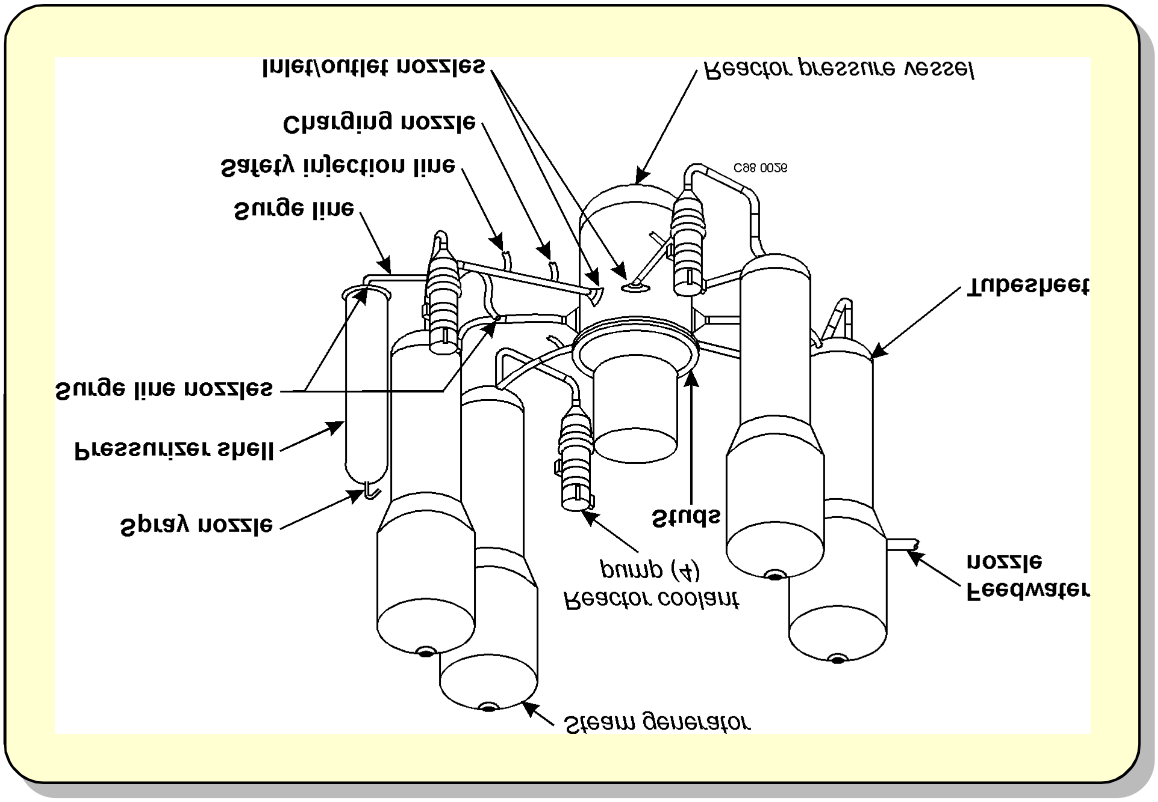

Figure 3-16. Schematic of Westinghouse PWR reactor coolant system. Locations of relatively high fatigue identified in bold print.

3.6.1 Current R&D on Piping, Pumps and Valves

Risk-based inspection models and technologies to support NDE are being developed and implemented. These methods can reduce the required number of ASME Section XI in-service inspection locations and frequency of inspections and still maintain safety by focusing inspections on the most risk-significant locations. They are expected to save each operating nuclear plant a minimum of $2 to $3 million each ISI ten-year inspection interval.

Work is being performed to improve the understanding of the effects of high-cycle fatigue on socket weld life, to improve socket weld fatigue resistance, and to develop practical methods to characterize socket weld quality in order to identify fatigue-sensitive locations. The work includes studies of fracture mechanics to help understand the effect of root pass defects on fatigue behavior, an NDE program to characterize the socket weld roots, a welding study to develop practices that will yield consistently "good" socket weld performance, and fatigue testing to qualify results.

Work is also being performed to optimize stress intensity factors and stress indices for current piping components, which are known to be overly conservative, and to develop new indices not presently available. It is expected that the current stress intensity factors will be reduced by at least 30 percent, which could reduce nuclear plant O&M costs by as much as $100,000 for each

18-month operating cycle. The results of this work will be included in a reference handbook and a computer program which will provide a simplified tool for the analyst to evaluate specific components.

FatiguePro™ is an on-line fatigue transient monitoring system that is being upgraded to address future NRC and ASME Section XI operating plant fatigue assessment standards. A handbook has been developed to provide screening criteria which may be used to identify fatigue-sensitive locations (thermal and mechanical) in nuclear and balance-of-plant piping. It also provides fatigue evaluation recommendations and cost-effective methods and guidelines to effectively control and manage fatigue in operating power plants. This handbook, software, and associated fatigue damage databases will be maintained on a continual basis.

A pilot study at Oconee Nuclear Power Station is being performed to demonstrate the continued integrity of Class I components that are subject to fatigue through the current license period and for potential license renewal consideration. Results of this project will help provide information necessary to address NRC concerns regarding the aging management of fatigue (including consideration of environmental effects) for license renewal for branch line piping and components. Fatigue analysis at Oconee will be completed in 1998. Application of revised stress intensification factors and consideration of environmental effects on Class 1 component analysis results will be completed in 1999. The pilot study, including the final EPRI report with generic application of results to PWRs for license renewal, will be completed in 1999. EPRI is also sponsoring projects to evaluate fatigue environmental effects for other PWR and BWR components.

Another current project deals with improving inspections of raw water and service water systems. Improved technology and training for inspection of balance-of-plant heat exchangers will reduce the frequency of leaks. Many heat exchangers are located in safety-related systems in nuclear plants, and their reliability is critical. Reliable inspection of BOP heat exchanger tubing is difficult because of the wide variety of alloys, sizes, and configurations in use and the many different forms of damage that have been found. Although eddy current techniques have been applied successfully for many years to steam generators and BOP heat exchangers, the effectiveness of specific eddy current procedures for a particular situation has not been systematically evaluated. Electromagnetic surface examination methods offer significant cost savings and improved capability relative to traditional liquid dye penetrant or magnetic particle inspection methods, particularly in radiation areas. In addition to tubing, EPRI is actively engaged in studying the application of eddy current techniques for many other plant components. A major initiative in this project is development and application of a cost-effective heat exchanger condition assessment program. The key features are selecting the most important components for inspection, selecting the appropriate inspection procedures, applying realistic tube plugging criteria, and trending the results of successive examinations.

3.6.2 R&D Needed on Piping Systems

The implementation of the risk-based inspection technologies being developed by EPRI and ASME will greatly reduce the cost of piping inspections and improve their effectiveness. However, further improvements in technology can help to reduce the costs and improve the effectiveness of inspections even further. Fatigue failures in small components are a significant cause of unscheduled plant shutdowns. The uncertainties associated with estimating fatigue damage in LWR coolant systems may be a source of regulatory difficulty in license renewal. Cast stainless steels, widely used in piping systems and for pump and valve bodies, are very difficult to inspect. Additional research is needed to find more effective inspection techniques for these materials. Managing flow-accelerated corrosion (FAC) also remains an expensive problem for utilities. There are on the order of 5000 components that could be affected by FAC. A typical utility FAC program may require detailed measurements on 50-200 components; development of innovative inspection procedures that could reduce FAC management costs is needed. The complex geometries and use of cladding make inspection of dissimilar metal welds very difficult. Eight tasks which address these issues are identified and discussed in Volume III . These tasks can be grouped into four categories: piping fatigue, NDE characterization of cast stainless steel components, flow assisted corrosion of piping and fittings, and inspection of dissimilar welds.

Fatigue. During the initial licensing period, very conservative assumptions on stresses and cyclic histories could be made and still demonstrate that fatigue damage will remain within acceptable limits. However, in the license renewal period it will be increasingly difficult to take this approach. The development of sophisticated monitoring programs by EPRI reduces the conservatisms associated with assumed fatigue loads, but is an expensive way to manage a problem that may be caused primarily by conservatisms in the calculated stresses and does not really reflect the real damage seen in the component. Research is needed to more realistically evaluate fatigue stress intensification factors at joints, the effects of thermal versus mechanical loadings, the effect of the environment on fatigue life, the actual vibrational fatigue life of socket welds, and to be able to characterize the real damage in a component.

NDE for Cast Stainless Steel. These tasks will improve the inspection and assessment capabilities for cast austenitic stainless steel piping and valves by application of advanced ultrasonic and signal processing techniques, a local stress-strain microprobe system (automated ball indentation testing), and magnetic techniques. Improved and faster NDE for cast stainless steel will reduce inspection intervals and costs of operating a nuclear power plant. The microprobe system will facilitate more accurate assessment of thermal aging damage to statically cast components.

NDE for Flow-Accelerated Corrosion and Dissimilar Metal Welds. Improved NDE techniques will be pursued to reduce the time and cost of detecting flow-assisted corrosion in nuclear reactor piping. This phenomenon involves loss of material over a large area, for example downstream from an elbow, rather than formation of cracks. While current ultrasonic techniques involving normal incidence longitudinal waves are capable of detecting flow-assisted corrosion, the task is arduous and requires the extensive removal of insulation. A system that will detect corrosion at a distance would have the distinct advantage of minimizing the cost of removing insulation from piping and reducing the time to locate the degraded region of the pipe. The in-service inspection capability for dissimilar welds also needs to be improved. Complex geometries and the use of cladding make traditional ultrasonic inspection of such welds very difficult. However, recent advances in modeling wave propagation in austenitic materials has made improved transducer designs possible.

The high priority projects identified for commencement in FY 1999 are listed below, along with the principal objectives of each project. See Volume II for detailed descriptions of these and other high priority projects in this area. See Volume III for descriptions of all projects in this area.

Project ID:	3-25
Project Title:	Fatigue
Principal Objective:	The purpose of this work is to provide cost effective methods of evaluating the cyclic life of nuclear components, including the effects of reactor coolant environment, based on the safety margins of the ASME Code.

3.7 Generic R&D

Much of the current research and development on aging management is component specific – such as the management of irradiation embrittlement effects on low-alloy ferritic steel reactor pressure vessel materials and the management of irradiation-assisted stress corrosion cracking of austenitic stainless steel reactor vessel internals. In some cases, however, R&D is more generic in nature, extending to vessels, tanks, piping, pumps, valves, internals, and balance-of-plant components. The crosscutting tasks described here are more generic in nature and affect all types of components. For example, some of the current R&D is focused on materials information and analytical assessment tools that are applicable to a variety of components. Other current generic R&D relates to work force quality and the application of modern risk assessment tools and technologies to capture the risk impact of aging. All of these topics have potential impact on the license renewal process.

In addition, with the availability of components and structural materials from plants that have been retired from service, generic data on the effects of aging through examination of components and structures can be obtained. This information can provide a useful, realistic assessment of aging effects on important plant components and will "benchmark" data on the effects of aging obtained from accelerated laboratory or test reactor studies.

Another example of a crosscutting task involves long-range research needed to develop and demonstrate methods for direct measurement and assessment of the early precursors of material degradation, such as in-grain or grain boundary precipitates and severe dislocation patterns in metals.

3.7.1 Current Generic Research

<u>Reduce Plant Inspection Costs</u>. The objective of a current EPRI project is to reduce plant inspection costs through the establishment of realistic NDE requirements and the introduction of advanced technology. This project supports integration of relevant EPRI research into the activities of industry groups such as the ASME Boiler and Pressure Vessel Code Group, ASME Pressure Vessel Research Council and NEI. Using EPRI results, ASME code rules and acceptance criteria will be developed and implemented for reduced RPV reference flaw size assumptions, risk-based inspection, environmental fatigue effects, optimized stress intensification factors, and flaw tolerance evaluations.

Risk-informed approaches to NDE are being developed for inspection applications, to optimize them and make them more cost-effective. New rules for selecting in-service inspection sites are being developed by ASME, the utility industry, NRC Office of Research, and EPRI. The sites selected under the new rules may not have adequate nondestructive inspection technology available to properly implement risk-informed inspections. This project provides technical input during development and implementation of risk-based inspection criteria. The 1998 focus is on supporting pilot plants and developing an NDE reliability model.

Another area of improvement under consideration is flow-assisted corrosion (FAC) inspection techniques, which provide a measurement of remaining pipe wall thickness. Eighty percent of current FAC inspection costs are associated with removal and reinstallation of piping insulation. An improved technology involving low-strength radiography for measuring wall thickness through insulation is being pursued by EPRI.

Another issue being addressed in this project is the examination of dissimilar metal welds (e.g., Alloy 182 safe-end welds). Some of those welds have such complex configurations that accurate interpretation of the inspection signals and photographs has been extremely difficult.

Advanced NDE Technology. Utilities need cost-effective and reliable inspection methods and analysis programs that can be integrated easily with structural and lifetime evaluations of nuclear power plant components and systems. This EPRI project focuses on the development of improved NDE hardware, software, databases, and methods. The deliverables address development of NDE technology to solve existing and projected integrity management issues while maintaining an element of innovative research to solve future problems. Nondestructive methods for addressing material properties and damage prior to cracking is included.

Materials Handbook. This research and development activity at EPRI is aimed at a materials handbook for nuclear plant pressure boundary applications that can be used as the primary source document by utility materials engineers facing questions regarding repairs, replacement materials, trouble-shooting failures, failure analysis, and in-service inspections. While much of the desired information is available in other publications, it has not been compiled in one place with emphasis on factors important to nuclear power plant pressure boundary applications. A related current research activity at EPRI is aimed at providing guidance on the useful service life of valve stem and bolting alloys that are susceptible to thermal embrittlement. Service failures have resulted from embrittlement of components in PWRs, possibly aided by stress corrosion cracking.

Equipment Assessment & Maintenance Technologies. The cost of operating and maintaining nuclear plants has created a serious challenge to the competitiveness of current and future plants. Cost-effective operations and maintenance (O&M) programs are essential to ensure the continued use of nuclear energy. The success of O&M programs depends on technologies that limit O&M activities, lead to less costly approaches to performing required work, deal with the problems of aging plants, make better use of human resources, provide more cost-effective instrumentation and controls, and support decisions to continue plant operation.

Utilities have reduced O&M costs by controlling staff size, improving plant performance, eliminating unnecessary activities, and applying least-cost options for required actions such as maintenance, inspection, surveillance and testing. Utilities will need to rely on new tools and continuous improvements in equipment reliability, assessment methods, and repair techniques to sustain the downward trend in O&M costs.

A current EPRI project addresses a broad set of plant and equipment reliability, assessment, and repair issues. It is producing specific solutions to these significant O&M cost drivers and is exploiting opportunities with cost-reduction potential. The deliverables will generally consist of 1) incremental design improvements to high-cost impact equipment which will significantly increase reliability, maintainability, performance and service life; 2) innovative lower cost alternatives for required maintenance, condition assessment, surveillance and testing; 3) defensible technical bases for justifying equipment operability and optimizing maintenance frequencies; and 4) technologies that will improve station output. This project includes major piping, valves, rotating equipment, stationary batteries, heat exchangers, and concrete structures, addressing many current and emerging industry issues. Strong coordination and resource leveraging with other organizations, both within and outside EPRI, is being maintained on this project.

Work Force Quality. The supply of qualified inspectors and their level of training and qualification is a concern being addressed by a current EPRI activity. Approximately 30 percent of the EPRI NDE program is directed to training of utility NDE inspectors. In spite of recent advances, training and qualification of NDE personnel continue to be characterized by high costs and low effectiveness. High costs are associated with the labor required to develop and implement training and qualification materials and for operators to reach acceptable proficiency levels. To meet these challenges, EPRI is developing improved procedures to increase reliability, validity, and effectiveness while also reducing costs. NDE training and qualification requirements that can benefit most from improvement will be identified. Interventions based on technologies most applicable to improving performance will be developed and tested. The methods and interventions being developed will support training and qualification programs, such as the Performance Demonstration Initiative.

3.7.2 Generic Research Needs

Section 3.7.1 identifies a number of issues and related work being pursued by EPRI. The scope of the currently planned work is limited and there is a need for expanding the scope and performing R&D to effectively and completely address these issues. For example, on the issue of work force quality, technology could be devised that creates a virtual environment, permitting an ultrasonic testing operator to conduct practice examinations while directly observing the entire process – effects of transducer manipulation, sound paths in the material, effects of geometry, displayed signals, and interactions. The immediate and dramatic feedback of the results of examination strategies, procedures and actions might significantly reduce the time required for operators to become proficient, and would provide a better foundation for skill retention.

There is a need to obtain data on the effects of aging through examination of components and structures that have aged under service conditions. Components removed from nuclear power plants are needed to provide a useful, realistic assessment of aging effects on important plant components and to "benchmark" data on the effects of aging obtained from accelerated laboratory or test reactor studies. In addition to providing information on the effects of aging, these components would also be useful in providing information on promising new techniques and devices for nondestructive material damage assessment (e.g., RPV "embrittlement meter" and direct measurements of fatigue damage). These components can be obtained from a number of sources, such as power plants which have been retired from service or components that have been replaced (e.g., steam generators). Opportunities to acquire such components for analysis in the recent past have been lost (e.g., Shippingport, Yankee Rowe, numerous steam generator replacements) due to 1) the unavailability of resources, 2) the lack of a systematic evaluation of the needs for component retrieval and assessment, and 3) the relatively short "window of opportunity" to obtain such components from the time a plant is shutdown or a component is replaced to the time the components are sent for disposal at a waste storage site or otherwise rendered inaccessible.

Long-term research is needed to develop and demonstrate additional methods of detection of material degradation, such as in-grain or grain boundary precipitates and severe dislocation patterns. The development of methods for direct measurement and assessment of early precursors requires the continued development of physically-based material models so that the characteristics of susceptible materials can be identified and quantitative predictions of damage or condition assessments as a function of service exposure can be made. Development of in situ examination techniques that can detect the physical manifestations of material degradation, correlating with the physically-based material models will also be needed. In-situ techniques for examining material composition, chemistry, and microstructure, leading to the identification of early indicators of material degradation, will first be demonstrated in the laboratory (or hot cell) and then validated through destructive examination of components removed from service.

The determination and quantification of nuclear plant risks from aging concerns will need to be addressed as plant owners prepare for license renewal. A comprehensive probabilistic safety assessment (PSA) package will allow plant owners and operators to prioritize maintenance, inspection, and replacement activities taking aging effects into account, and to address regulatory concerns related to public risk. Development of such a PSA package is another generic R&D need related to power plant aging.

The high priority projects identified for commencement in FY 1999 are listed below, along with the principal objectives of each project. See Volume II for detailed descriptions of these and other high priority projects in this area. See Volume III for descriptions of all projects in this area.

Project ID:	3-27
Project Title:	Assessment of Aging Effects on Components and Structures from Nuclear Power Plants
Principal Objective:	Obtain materials and components that have been in service in operating reactors to be used for comparison with laboratory aged materials to validate models for aging effects and nondestructive examination methods.