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New Nuclear Plant Design and Licensing Process

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1.0 INTRODUCTION

Mechanical or electrical failures have been the main reason for malfunctions at commercial nuclear plants. Under special circumstances, a malfunction may propagate into an accident. However, as we have seen in the past severe nuclear plant accidents such as at TMI-2 and Chernobyl-4 were very commonly caused by operator errors that were the direct results of insufficient training. Current nuclear power technology relies heavily on operators to follow appropriate procedures to respond to symptoms of a malfunction. One mistake committed may lead to an uncontrollable situation.

Fortunately, an improvement is always possible. A great deal has been learned about how to design a better and safer reactor. The technology of advanced passive light water reactors has progressed to the point that operators are not required to take actions during an accident. The new technolgy tremendously eliminates the role of human operator in bringing the plant to a safe state. The new plant design can bring itself to a safe stable condition following an abnormal event for up to three days without human interaction. After three days only simple operator actions are required to maintain the plant in a safe stable condition.

Advances in technology enhance safety to the level higher than never before. However, the real safety is not only determined by technology, but also by management, accountability and performance of an organization that operates and maintains the plants as well as those of the regulatory agency from the very beginning of licensing processes through the processes of plant construction and regulatory inspection until day-to-day operations of the plant.

In the lecture, we will focus our attention to the U.S. technology of advanced passive light water reactors, and the regulatory processes that certify its safety. The technology looks attractive to countries where the first nuclear power plant project is being considered and as such no operation experience exists. The goal is to get an insight into how new passive safety features were designed and tested by vendors and how they are re-evaluated and retested (i.e., independent check) by the regulatory agency. We will begin by looking into details of passive safety design features, then move on to see how the new nuclear licensing processes work in the U.S. and how they are applied to the new designs. At the end, we will look at some of generic engineering concerns expressed by the USNRC for current light water reactors related to material aging and corrosion that may be improved but not go away in the new design.

2.0 PASSIVE SAFETY DESIGN OF NUCLEAR PLANTS

As of now, there is only one passive design that is being certified in the U.S.. This design is the Westinghouse AP600. The design certification processes are nearing the end. It is expected that the AP600 design will receive a Final Design Approval (FDA) from the USNRC by early 1997. After that point, it will become commercially available for overseas markets. General Electric cancelled its passived design SBWR program early 1996 due to sponsor cutbacks, and extended design certification cost.

2.1 AP600 Design

The AP600 primary system is a two-loop design (refer to Figure 2-1). Each loop contains one hot leg, two cold legs, and one steam generator with two canned reactor coolant pumps (RCPs) attached directly to the steam generator outlet channel head. The passive safety systems comprise the following:

- Two full-pressure core makeup tanks (CMTs) that provide borated makeup water to the primary system at any pressure.

- Two accumulators that provide borated water to the reactor vessel when/if primary system pressure $\leq$ 700 psia.

- A passive residual heat removal (PRHR) heat exchanger (HX), comprised of a C-shaped tube bundle submerged inside the in-containment refueling water storage tank (IRWST), that can remove heat from the primary system at any pressure.

- The automatic depressurization system (ADS), which is comprised of a set of valves connected to the pressurizer steam space and the two hot legs. These valves are opened sequentially to provide a controlled depressurization of the primary system.

- An IRWST that provides a large source of core cooling water, which drains by gravity after the ADS has actuated.

- A passive containment cooling system (PCS) (refer to Figure 2-2) that utilizes the AP600 steel containment shell to transfer heat to the environment (ultimate heat sink).

In reviews of the AP600, the U.S. Nuclear Regulatory Commission (NRC) identified several concerns regarding the performance of the AP600 passive safety systems. Those concerns include the following:

- Possible high-pressure passive safety system interactions which could retard cooling to the core.

- The dependence on small temperature differences resulting in small density differences, which then are responsible for driving heads for recirculating flows.

- The effects of code accuracy in predicting long transients in which the driving heads for the flows in the systems are small.

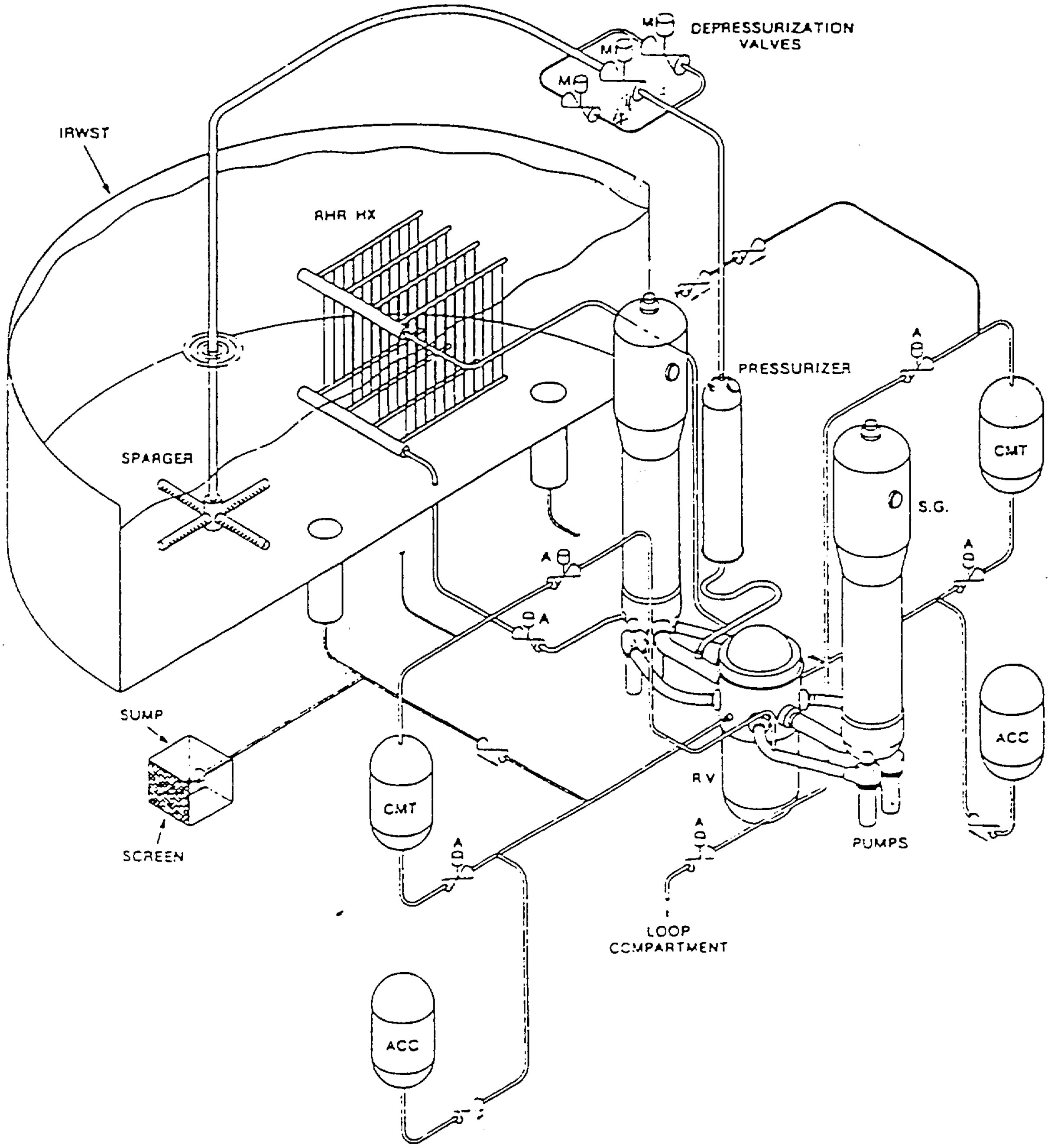


Figure 2-1 AP600 Passive Safety Systems.

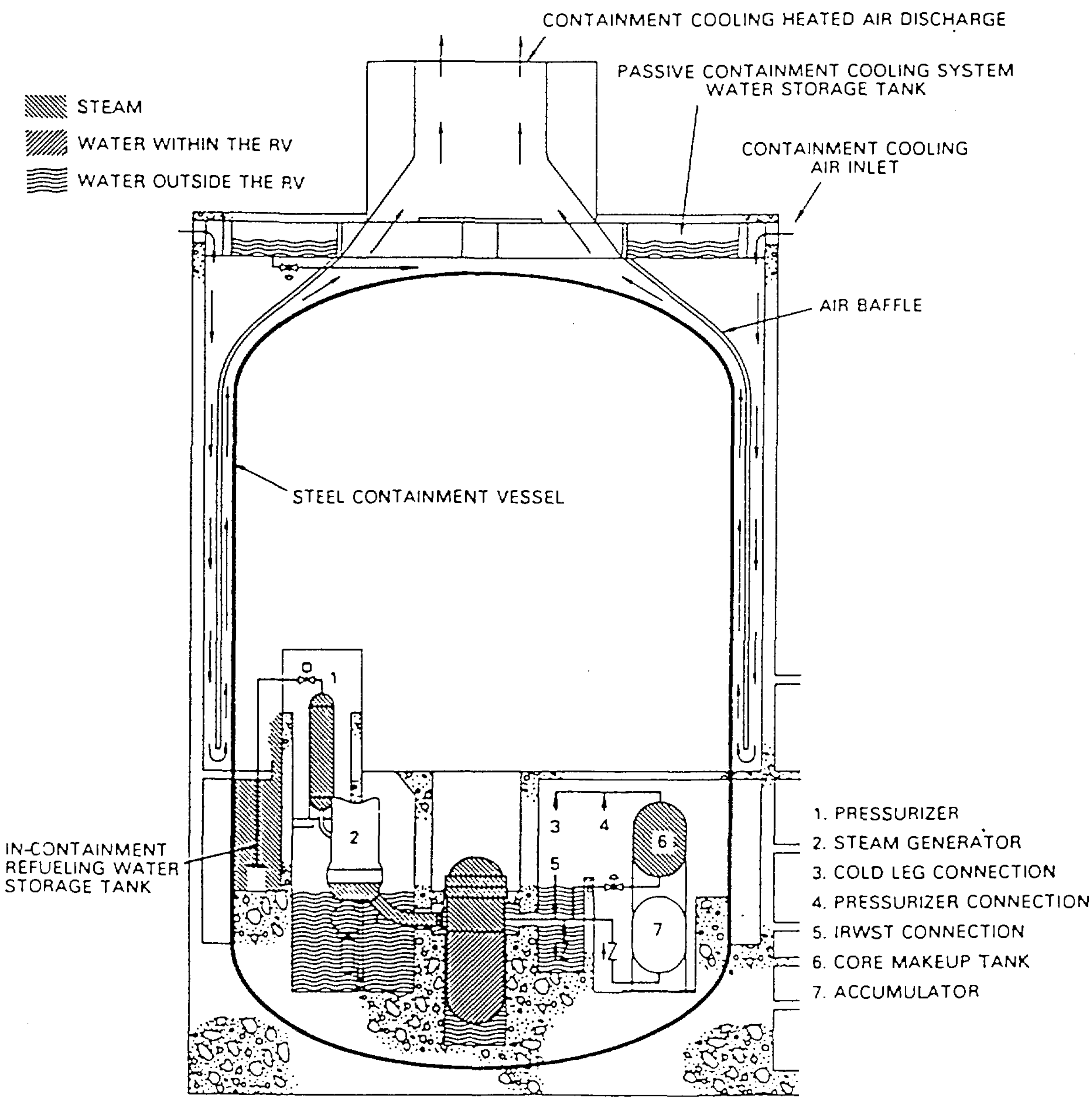


Figure 2-2 AP600 Passive Containment Cooling System.

To address the NRC concerns, tests were conducted in three general categories: component design verification tests, passive containment cooling system tests, and passive core cooling system tests. The test program was completed at the end of 1994. By then, the AP600 became the most thoroughly tested advanced passive reactor design ever reviewed by the USNRC. Very sophisticated, detailed, conservative computer codes have been validated for analyzing the AP600. The final design approval of the AP600 is expected by the end of 1996 or early 1997. While the NRC's review of the test program data is still being completed, the NRC has acknowledged the simplicity and safety of the design (Bruschi, 1996).

Key Thermal-hydraulic Phenomena: A Small Break LOCA

Passive safety system is an innovation in thermal hydraulic design. The key to understand how safety systems are designed is to understand the thermal-hydraulic phenomena of the system under abnormal conditions. Here, a small break LOCA phenomena are described as an example (Wright, et al., 1996).

For the blowdown phase, during a small break LOCA, the break opens and the pressurizer pressure begins to fall as mass is lost out the break. This depressurization is largely defined by critical flow through the break. With the break located at the bottom of the cold leg, a mixture flow exits the break for the majority of the transient, since the mixture level stays high in the reactor vessel. The pressurizer pressure falls below the safety signal setpoint, causing the reactor to trip. The safety systems actuation (S) signal follows and results in opening the CMT and PRHR isolation valves. Once the residual fissions decrease, core power is defined by the decay heat model. The RCPs trip after a short delay and the rapid coastdown is expected from the AP600 canned motor RCPs. After the pumps coast down, the primary reactor coolant system (RCS) is cooled by natural circulation, with energy removed from the primary system by the steam generators, via their safety valves, and the break. Stored energy from the metal in the primary system is transferred to the coolant. Note that all of these phenomena are essentially the same for AP600 as for conventional PWRs. The liquid in the upper plenum and upper head (depending on the temperature) will flash, and upper head start to drain.

Blowdown phase phenomena unique to the AP600 are those associated with the CMT delivery. Once the CMT isolation valves open on an S actuation signal, the CMT injects borated water due to gravity-driven recirculation into the RCS through the DVI lines. The CMT volume injected is replaced with hot liquid from the cold-leg, which circulates through the cold-leg balance line (CLBL); this hot liquid collects at the top of the CMT. The downcomer fluid stays subcooled through the initial depressurization.

For the natural circulation phase of the transient, the primary system exists in a quasi-steady-state condition with the secondary side, with the decay energy being removed by the steam generator secondary side and the PRHR, as the primary system drains. The steam generator in AP600 plays a more limited role in the natural circulation cooling phase than for conventional plants because the generators' primary sides drain relatively early in the transient. Since the PRHR is activated early during an SBLOCA, after an S signal is issued, the IRWST becomes the primary heat sink for the RCS early in the transient; the PRHR will remove energy from the primary system, causing it to further depressurize. The CMT continues to deliver in the recirculation mode for a while, but eventually a vapor region forms at the top of the CMT volume and CMT draindown begins. As the CMT drains while injecting, its level falls to the ADS actuation setpoint, which initiates the third phase of the AP600 SBLOCA transient, the ADS phase.

The ADS phase continues through ADS-1, ADS-2, ADS-3 and ADS-4 as the primary system completely depressurizes approximately to the containment pressure. Since ADS-1 creates an opening at the top of the pressurizer, the pressurizer two-phase fluid level increases markedly. Pressurizer tank level and surge line phenomena are significant factors in the depressurization behavior following ADS actuation. Flashing of fluid in the RCS occurs again due to the depressurization caused by the ADS.

Following ADS-1 actuation, ADS-2 and ADS-3 activate in sequence via pre-set timers. Accumulator injection begins once the pressure drops below 700 psia which reduces the flow delivered from the CMT, and CMT flow may even be stopped temporarily due to pressurization of the DVI line by the accumulator. The CMT drain rate, DVI line and CLBL behaviors are significant because the ADS-4 actuation is based upon the CMT liquid level decreasing below a low-low setpoint value.

Critical flow through the ADS stages is the major factor in determining when the RCS has depressurized to the extent that the gravity injection of water from the IRWST can begin. Fourth-stage ADS performance will be particularly affected by the nature of flow in the hot legs. Successful operation of the ADS leads into the IRWST injection cooling phase of the AP600 SBLOCA event.

The final stage of the SBLOCA is IRWST injection. At this point, the primary system is completely depressurized and the transient continues into the long-term cooling phase of the accident. By the time of IRWST injection, the CMT is either completely empty or very nearly empty; therefore, CMT phenomena have become relatively unimportant, whereas the IRWST gravity drain rate through the DVI is important. The hot-leg flow phenomena, together with the ADS-4 flow, is also important. Moreover, the break critical flow behavior is now less important than before because all the ADS flow paths are open, providing a large area through which to vent steam. Maintaining the core covered with liquid or a two-phase mixture becomes a function of the decay heat level and the IRWST flow.

The key thermal-hydraulic phenomena for the non-LOCA transients include fluid flashing in the reactor heated, forced and natural circulation flow in the primary system, steam generator heat transfer, CMT recirculation and the pressure drop and temperature distribution in the balance lines. Computer codes that are being used to analyze the AP600 design have been validated based on the various testing programs. The data and codes predict safety under a long list of scenarios, both normal and abnormal. The AP600 has been shown by computer simulations to be safe under all predicted accident conditions.

2.2 SBWR Design

The SBWR is a 670 MWe power plant. The plant uses a simple direct-cycle, natural circulation boiling water reactor. Sixty percent of the SBWR systems are identical to or similar to ABWR systems (Upton, et al., 1996). The main differences from previous containment designs are: 1) the location of the Suppression Pool at an elevation above the core and thus positioned above the reactor building basemat to allow long term core cooling by gravity drain and 2) the incorporation of all ECCS coolant inside the containment boundary in three individual (and thus divisionally separated) Gravity Driven Cooling System (GDCS) pools (refer to Figure 2-3).

The key feature of the SBWR passive safety design that differentiates it from other ALWRs is how the SBWR responds to a design basis event such as a loss-of-coolant-accident (LOCA) and the time that the operator has to respond to the event.

Following a LOCA in SBWR, all safety related functions are provided by passive means. Passive means for SBWR consist of natural forces such as gravity, natural circulation, stored energy such as batteries and compressed fluids, check valves, and non-cycling powered valves.

The SBWR reactor response and containment response following a LOCA can be divided into three phases:

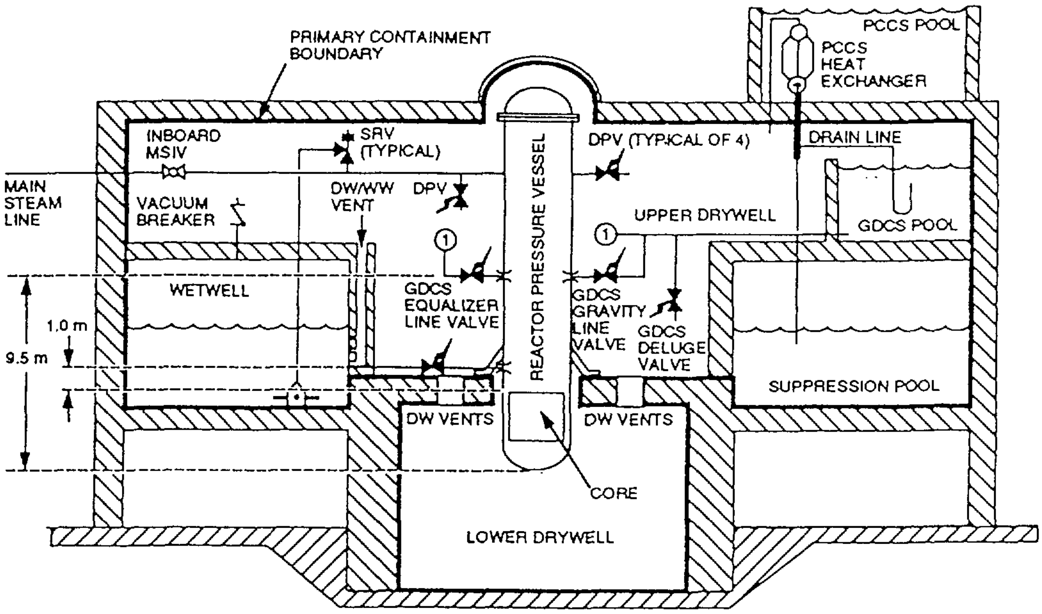
- Reactor pressure vessel blow-down phase where the RPV depressurizes, enabling the GDCS to start injecting water into the reactor vessel;
- GDCS phase during which the GDcS pools drain into the RPV, and
- the long term cooling phase after the GDCS pools have drained and the Passive Containment Cooling System (PCCS) removes decay heat and recycles condensed steam to the RPV.

Since every LOCA event rapidly results in vessel blowdown via the Automatic Depressurization System (ADS), the predicted SBWR containment response is very similar in each type of LOCA event.

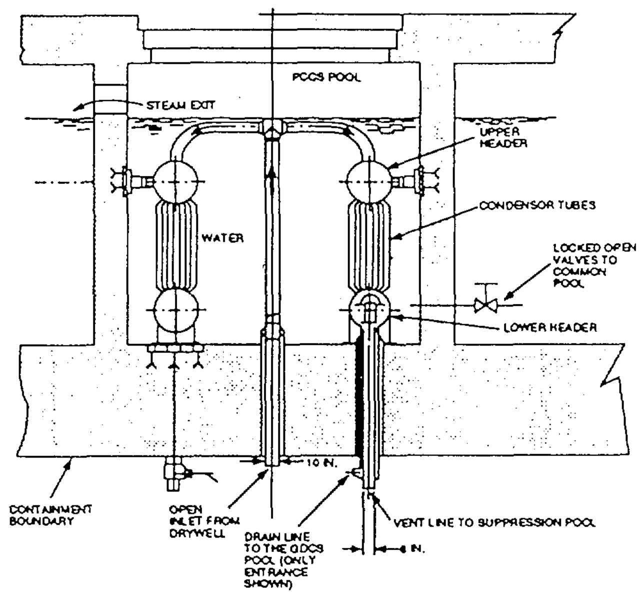
The SBWR conceptual design is complete. Significant progress has been made on plants detailed design (Upton, et al., 1996). Significant component and qualification testing has been completed on key portions of the design and on integrated SBWR response to a LOCA with favorable results. However, as mentioned earlier, the SBWR program was cancelled. GE will not seek design certification and will not market the SBWR. GE redirects resources to plants of 1000 MWe or larger, including the ABWRs.

3.0 NUCLEAR LICENSING REFORM IN THE U.S.

In April 1989, to enhance nuclear plant safety and make the regulatory approval process more predictable and less costly, the NRC issues new regulations (i.e., 10 CFR Part 52) with emphasis on standardized plant designs providing for early and final resolution of site, design, and construction issues. The policy objectives of 10 CFR Part 52 are to promote the standardization of nuclear power plants and provide for the early and definitive resolution of site and design issues before plants are built. Thus, Part 52 moves the bulk of the licensing issues, and hearings on them, to the front of the process, while reserving for later proceedings only



SBWR CONTAINMENT AND PCCS, ADS, AND GDCS ARRANGEMENT



OUTLINE DRAWING OF PCCS HEAT EXCHANGER

Figure 2-3 SBWR Passive Safety Design.

those construction related issues that cannot be resolved at the front end (Fox & Quirk, 1996).

The most innovative licensing feature of Part 52 is the new public hearing procedure that would precede authorization of facility operation. Under the old (Part 50) licensing process, the operating license proceeding was open to a full-scale contest of virtually all design and site issues. This was one of the most compelling reasons for the unpredictability of NRC reactor licensing and a major reason for utility and financial community disenchantment. In contrast, the new (Part 52) regulations confine pre-operational hearings to issues of non-compliance with pre-established acceptance criteria, the satisfaction of which is determined by completion of pre-established inspections, tests and analyses. Requests to reexamine issues previously settled, on the basis of alleged new information, can only be made subject to discretionary Commission approval. Moreover, under authority granted by the Energy Policy Act of 1992, pre-operational hearings can be informal and the Commission can authorize interim facility operation pending hearing completion if it finds there would be no safety impediment to such operation.

Part 52, illustrated in Figure 3-1, establishes procedures to obtain early site approvals, pre-approval of standardized designs, and the issuance of combined construction and operating licenses (COL), along with a process for pre-operational authorization that accords finality to matters covered by the prior regulatory approvals. Each stage of the new licensing process is accompanied by a hearing opportunity which affords the affected public the ability to be heard in the decision-making process. There is, it is believed, a wide-ranging consensus that a three-pronged approach--early site approval, design standardization, and issuance of a combined license prior to commencement of construction--provides the best basis for solid and productive reform.

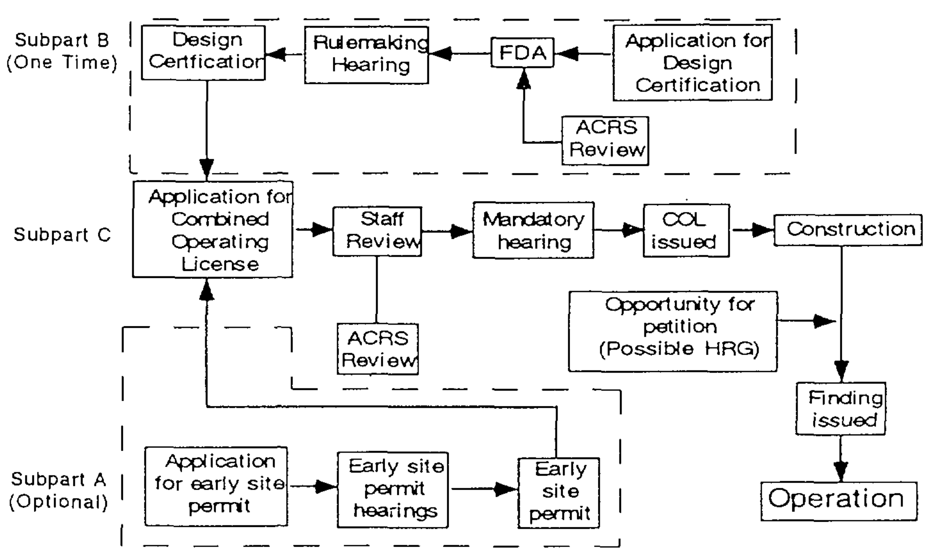


Figure 3-1 10 CFR 52 Licensing Process in the U.S. (from Fox & Quirk, 1996)

3.1 Early Site Approval

Part 52 provides the procedures to obtain approval of sites for nuclear power plants in advance of the filing of an application for plant construction on such sites. These so-called early site permits (ESP) are conditional on future plants fitting within an envelope of general design parameters and meeting specified population density and other site-specific conditions. An ESP approves a site for one or more nuclear power plants separate from the filing of an application for a combined construction permit and operating license (COL). Under Part 52, an ESP referenced in a subsequent application for a COL insulates from regulatory reconsideration all matters resolved in the early site proceeding long before the process of licensing a specific new power reactor begins. An ESP remains valid for up to 20 years from its issuance date, and is renewable.

In addition, the ESP proceeding can be the forum for resolving emergency planning issues. In such cases, the application for an ESP would identify physical characteristics unique to the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans. The application may also propose major features of the emergency plans, such as the exact size of the emergency planning zones, or may propose complete and integrated emergency plans for the site for review and approval by NRC in consultation with Federal Emergency Management Agency (FEMA). The application must include a description of contacts and arrangement made with local, State, and Federal governmental agencies that have emergency planning responsibilities. Prior to issuing an ESP, the NRC must conduct formal adjudicatory proceedings covering all of the issues involved. The hearing provides affected persons the opportunity to raise most site-related issues. Early hearings on emotionally charged issues and resolution of these issues allows utilities (or other site permit applicants) a greater degree of certainty in planning future nuclear power plant projects.

3.2 Design Certification

Procedurally, following the issuance of a Final Design Approval (FDA) by the NRC Staff, NRC approval of a design certification is accomplished through rulemaking proceedings. Issuance of the FDA culminates the major portion of the technical review by the NRC Staff and ACRS. The FDA is a prerequisite to design certification. The design certification rulemaking process is initiated when the NRC Staff publishes in the Federal Register a notice requesting public comment on the content of the proposed design certification rule (DCR) that is published together with the notice, and on specific issues related to development of the DCR and its implementation in the Part 52 licensing process. Members of the public may provide written comments to the NRC and avail themselves of the opportunity to request an informal hearing before an Atomic Safety and Licensing Board (the "Board") on specific issues. The Board's role, if the Commission grants a hearing, is to assure compilation of an adequate record for Commission decisions on issues raised by parties to the rulemaking hearing. The Board is not to make a separate review of the application, nor is it to submit its own recommendations on disputed issues to the Commission. The Board may request additional authority from the Commission to allow cross-examination by the parties to the hearing, allow discovery by the parties or recommend that the Commission convene a formal hearing. A formal hearing, however, will be held only if it is required to resolve specific and substantial issues of fact, necessary to the Commission's decision, that otherwise cannot be resolved with sufficient accuracy.

If no requests for an informal hearing are made, the NRC will prepare a final rule for each design based upon its respective rulemaking record. That record would include the technical basis for the proposed rule, including the application for certification and its amendments during the staff review process, the two-tiered Design Control Document (DCD), the NRC Staff's Final Safety Evaluation Report (FSER) issued in support of the FDA, and the comments received on the proposed rule. After completion of the rulemaking process and a Commission determination that the application meets the applicable standards and requirements of the Atomic Energy Act and the NRC's regulations, the Commission will issue a standard design certification in the form of a rule. Design certifications have a duration of fifteen (15) years and are renewable, by rulemaking, for a period of not less than ten (10) or more than fifteen (15) years.

3.3 Licenses to Construct and Operate

In contrast to the two-stage construction permit and operating license proceedings of the former licensing process, the new regulations provide for a combined construction and operating license. The combined license proceeding considers only those issues which were not resolved in the proceedings for a site permit or a certified design referenced in the license application. Previously resolved issues may not be reopened. The interface between design and site, emergency planning, descriptions of operational programs and the qualify assurance program, pre-operational and initial test plans, security and safeguards plans, organizational structure and personnel qualifications, a full environmental impact statement, and any proposed ITAACs that apply to construction of the plant, comprise the matters that must be addressed in the combined license proceedings. In other words, the technical information requirements for a combined license essentially fill the information gaps left after the design certification and site approval processes.

The COL application must include a Final Safety Analysis Report (FSAR) that will provide the technically relevant information identified above. A COL application may reference a design certification rule. In this case, the Design Control Document (DCD) of the design certification rule will be referenced by the FSAR, thereby reducing substantially the amount of design information and ITAAC that would otherwise be required as part of the FSAR. Additionally, the FSAR will have to demonstrate with the site parameters and interface requirements in CDM of the DCD, and will have to address the COL License Information identified in Tier 2.

Prior to issuing a combined license, the NRC must hold a formal adjudicatory hearing and find upon completion of the proceeding that relevant regulatory requirements have been met and that there is reasonable assurance that the facility will be constructed and operated in conformity with the license, the Atomic Energy Act and the NRC's regulations. In that connection, the NRC must identify (in the license) the inspections, tests, and analyses that the licensee shall perform and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurances that the facility has been constructed and will be operated in conformity with the license, the Atomic Energy Act and the NRC's regulations.

3.4 Pre-Operational Verification

As discussed above, an integral part of the new licensing reform regulations is "front-loading" the resolution of licensing issues, i.e., resolving all design and siting issues before plant construction begins, leaving only verification for pre-operational determination. To achieve this, a mechanism was devised to enable NRC verification that the as-build plant conforms to the design and other requirements addressed in the earlier stages of the regulatory process. That mechanism also defines the issues that would be appropriate for regulatory decisionmaking and public participation at the pre-operational stage of the licensing process. The mechanism devised--and the truly unique feature of Part 52--are the inspections, tests, analyses and acceptance criteria ("ITAAC") incorporated in a design certification (and, as supplemented, in a referencing combined license) that are "necessary and sufficient" to provide reasonable assurance that, if they are satisfied, a plant which references the design is built, will operate in accordance with the combined license, the Atomic Energy Act and the applicable NRC regulations. ITAAC are thus created for the purpose of ensuring that the design was faithfully implemented and limiting pre-operational hearings to issues of ITAAC compliance. The ITAAC are the key ingredient of Part 52, connecting the design certification, combined license and pre-operational proceedings. Ultimately, it is satisfactory performance of the inspections, tests and analyses and compliance with the acceptance criteria that are the sole bases for determining whether the fully constructed facility will be permitted to operate.

3.5 Inspection During Construction

As indicated, the NRC must assure that the required inspections, tests, and analyses are performed, and prior to operation the Commission must find that the prescribed acceptance criteria are met. At appropriate intervals during construction, the NRC Staff will publish Federal Register notices of the successful completion of inspections, tests, and analyses under a so-called sign-as-you-go (SAYGO) process.

Under the NRC's Part 50 inspection and enforcement program, the NRC will conduct inspections and audits as part of its normal regulatory activities to ensure licensee compliance with the COL requirements and NRC regulations (Figure 3-2). Noncompliance with a requirement (including the requirement in Tier 2) would be subject to Part 50 NRC enforcement action.

3.6 Pre-operational Hearing

Not less than one hundred and eighty (180) days before the date scheduled for initial loading of fuel into a plant by a COL licensee, the Commission will publish in the Federal Register a Notice of intended operation. That notice provides that any person whose interest may be affected by operation of the plant may, within sixty (60) days, request the Commission to hold a hearing on whether the facility as constructed complies, or on completion will comply with the acceptance criteria of the license.

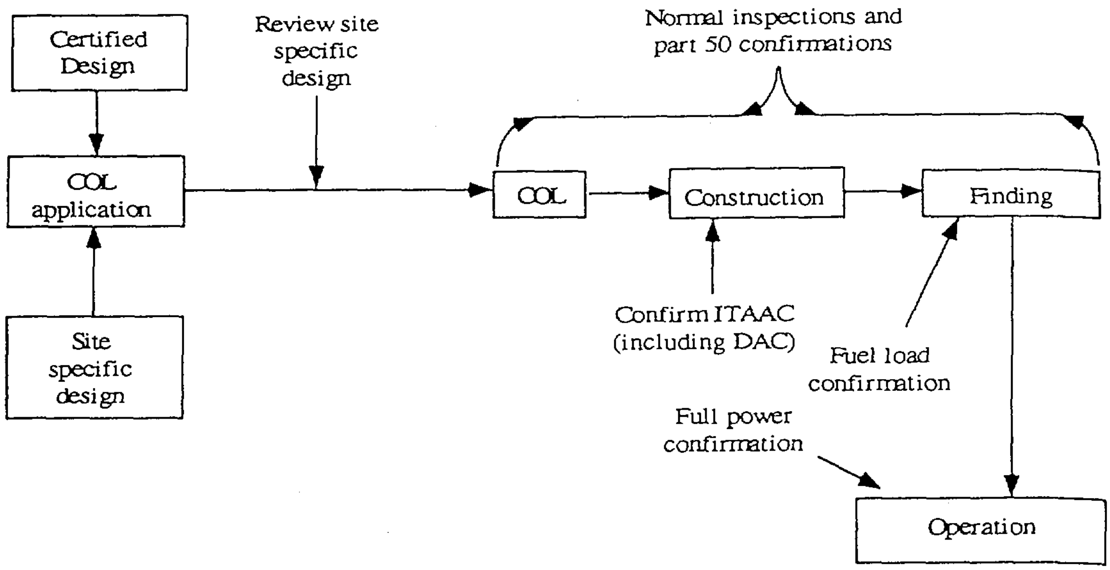


Figure 3-2 NRC activities during COL review. Construction and operation (from Fox & Quirk, 1996).

4.0 THE USNRC APPROACH TO DESIGN CERTIFICATION OF NEW NUCLEAR PLANTS WITH PASSIVE SAFETY FEATURES

As of December, 1996, the only advanced passive design being certified by the USNRC is the Westinghouse AP600. General Electric pulled out its SBWR from the certification process in 1996 due to escalating cost associated with uncertainties in the certification acceptance criteria.

The advanced Passive-600 (AP600) is a new pressurized water reactor (PWR) design of the Westinghouse Electric Company (WEC) that has been submitted to the United States Nuclear Regulatory Commission (USNRC) for design certification. For its safety, the AP600 relies on operation of passive systems unique to this new generation nuclear power plant (NPP). Within the certification process, the USNRC will perform selected system response audit studies to help confirm parts of the vendor's safety analysis submittal. Because of certain innovative design features of the safety systems, new experimental data and related advances in the system analysis computer codes are being developed by the USNRC.

The focus of the USNRC research is on those accidents that activate or challenge the new passive safety systems, and/or produce new interactions among those safety systems. The USNRC audit studies related to large break loss-of-coolant accidents (LBLOCA) are being performed with the TRAC-PWR system analysis computer code by the USNRC and Los Alamos

National Laboratory (LANL) staff (Boyack, 1995). Other transient analyses are addressed using the RELAP5 system analysis computer code and are performed by USNRC and INEL staff. These accident sequences include small break LOCA (SBLOCA), main steam line break (MSLB) and steam generator tube rupture (SGTR).

The phenomena identification and ranking table (PIRT) process, a part of the code scaling, applicability and uncertainty (CSAU) evaluation methodology, is being used to focus the experimental and analytical work to obtain a sufficient and cost effective research effort.

The PIRT process focuses on phenomena/processes that are important to the particular scenario, or class of transients, in the specified NPP (i.e., those that drive events). Plausible physical phenomena and processes, and their associated system components are identified first. These are then ranked with respect to their influence on the primary evaluation criteria to establish PIRTs. The identification and ranking are justified and documented. Because the dominant processes change during the transient progression, the scenario is normally partitioned into time phases and each phase is separately investigated. The processes and phenomena associated with each component are examined as are the inter-relations among the components. Cause and effect are differentiated. The processes and phenomena and their respective importance (rank) are judged by examination of experimental data, code simulations related to the plant and scenario, and the collective expertise and experience of the evaluation team.

The final product of application of the PIRT process is a set of tables (PIRTs) documenting the ranks (relative importance) of phenomena and processes, by transient phase and system component. Supplemental products include descriptions of the ranking scales, phenomena and processes definitions, evaluation criteria, and the technical rationales for each rank.

4.1 AP600 Case

Application of the PIRT process have provided the technical rationale to focus the USNRC research associated with the experimental program and subsequent code improvements. This information was needed to provide technically sound safety audit tools for use in the AP600 design certification in a cost effective manner (Wilson, et al., 1996). Figure 4-1 illustrates the central and key role of the PIRT process in the development of the design certification audit tools.

The PIRTs are based on information originating in AP600 specific experiments, the integrated analyses of the data from these experiments, scaling analyses related to the experimental facilities and the plant design, and sensitivity calculations related to computer code simulation of the plant. The experimental program includes separate effect tests (SET) in the PRHR (Corletti, 1992), CMT (Cunningham et al., 1994) and ADS (Yeh et al., 1995) facilities. Integral effect test (IET) data are centered in the SPES (Rigamonti, 1994), ROSA (JAERI, 1995), and OSU (WEC, 1994) facilities. The integrated experimental data analyses include test

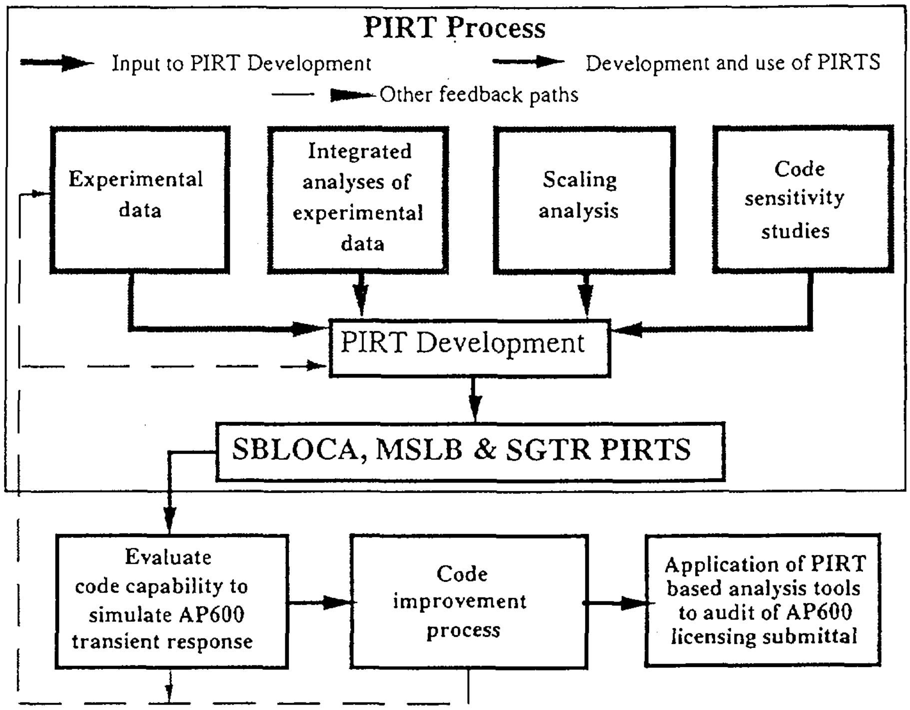


Figure 4-1 Role of PIRTs in the development of audit tools for the AP600 design certification (from Wilson, et al., 1996).

series related to cold leg breaks, direct vessel injection (DVI) line breaks, pressure balance line breaks, MSLB and SGTR transients. The scaling analyses are directed toward understanding the degree of prototypicality of the experimental facilities as related to the AP600 plant design, so that the data may be compared as a whole. Sensitivity studies are and will be directed toward determining the effects of one-by-one variations in highly ranked phenomena on the plant response for reasonable ranges of uncertainty in these parameters.

At the end of each stage in the PIRT development, the code modeling requirements, as represented by the PIRTs, are compared to the capability of the code to provide the modeling needs. Discrepancies between the code's modeling ability and the PIRT-based requirements then form the basis for continued code improvement. It may be noted the "discrepancies" are cast in terms of the relative importance of the PIRT-based requirements. That is:

- High rank = Phenomenon should be explicitly and accurately modeled,
- Medium rank = Phenomenon should be well modeled; accuracy may be somewhat compromised,
- Low rank = Phenomenon should be represented in the code, but almost any model will be sufficient.

Similar to the above evaluations related to code improvements, the PIRTs also provide the technical basis for judging the sufficiency of the experimental data base.

For simplicity, accident scenarios are partitioned into time phases during which plant behavior is dominated by the same processes. Figure 4-2 illustrates this procedure with an AP600 SBLOCA example.

PHASE	HIGH PRESSURE	ADS BLOWDOWN	LONG-TERM IRWST	LONG-TERM SUMP
TRANSIENT DURATION				
SIGNIFICANT PHASE FEATURES	Subcooled blow-down Reactor & RCS trip PRHR actuation CMT recirculation & draining	Completion of CMT draining Accumulator injection Transition to ADS stage 4 subsonic flow	IRWST injection to primary system	IRWST & sump injection to primary system
SIGNAL RELATED ELEMENTS		ADS actuation (CMT level at 67.5%)	Initiation of IRWST injection	Initiation of sump injection

Figure 4-2 Phases for AP600 SBLOCA PIRT process (from Wilson, et al, 1996).

5.0 ENGINEERING CONCERNS IN CURRENT LWRs FROM THE USNRC PERSPECTIVE

The advanced light water reactor design differs from the current generation in its improved safety system, i. e., the emergency core cooling system (ECCS) which is mainly an innovation in thermal hydraulics design, and in the use of digital technology in instrumentation and control (Chiramal, et al., 1995). The advanced design and the current generation are similar in all other aspects. Both use water as coolant. Both use similar vessel and piping materials that may be differed by slow progress in material technology over time. These materials will be subject to corrosion at various degrees, regardless of where they are being used, in the new design or in the current design. Of course, the new design will always use the best materials known at the time it is built. The following are some of the generic concerns expressed by the USNRC (Sheron, 1995). The concerns are related to both BWR and PWR. These concerns emphasize the importance of material and water chemistry technology in dealing with corrosion problems in water-cooled nuclear reactor systems.

In addition, the NRC also raised a concern on motor-operated valves (MOV) used in the safety system of current plants. In the new passive safety design, the system will work as designed only if valves, required to open, open as they are supposed to. Although there are 60% fewer valves in the new design than in the current design, this concern should also apply to the new design.

5.1 Cracking of BWR Reactor Internals

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

Cracking of the BWR core shrouds and jet pump hold-down beams (refer to Figure 5-1) have been the most significant of BWR internals cracking reported in 1993 and 1994. As operating BWRs age, the number of cracking incidents is expected to increase. The USNRC staff has been meeting every year since 1988 with the Boiling Water Reactor Owners Group (BWROG) and the General Electric Company (GE) to review the generic safety implications of reactor internals potentially susceptible to IGSCC. In a January 1992, meeting the staff emphasized to the BWROG that a comprehensive program should be developed to proactively address internals cracking and that utilities should adopt an enhanced inspection program.

5.1.1 Core Shroud Cracking

The core shroud is a stainless steel cylinder which partitions feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core. Cracking of the core shroud was visually observed in 1990 in a European BWR and was located in the heat affected zone of a circumferential weld in the beltline elevation of the shroud. Initially, a number of U.S. BWR licensees performed visual examinations of their core shrouds in accordance with the recommendations of General Electric. During their spring 1994 outages, significant cracking was found near some of the welds in the core shrouds at two Mark I BWR/3 units.

In light of the extent of the cracking observed at the two Mark 1 BWR/3 units, the staff evaluated potential safety concerns associated with the possibility of a 360° circumferential separation of the shroud following a pipe break. Such separation of the shroud might either prevent full insertion of the control rods or open a gap in the shroud large enough for the resulting gas leakage to preclude adequate core cooling by the emergency core cooling system (ECCS). The accident scenarios of primary concern are the main steam line break and the recirculation line break, normally referred to as loss-of-coolant-accidents (LOCAs). The more serious event associated with cracks in the upper shroud welds (e.g. H2, H3) is the steam line break, since the lifting forces generated may be sufficient to elevate the top guide and potentially cause difficulties with rod insertion. The more serious event associated with cracks in the lower elevations of the core shroud is the recirculation line break. This is because for the lower welds (e.g., H4, H5) the recirculation line break could potentially cause a lateral displacement of the shroud or opening of a crack that could allow enough leakage through the shroud and out the break to prevent adequate cooling of the core.

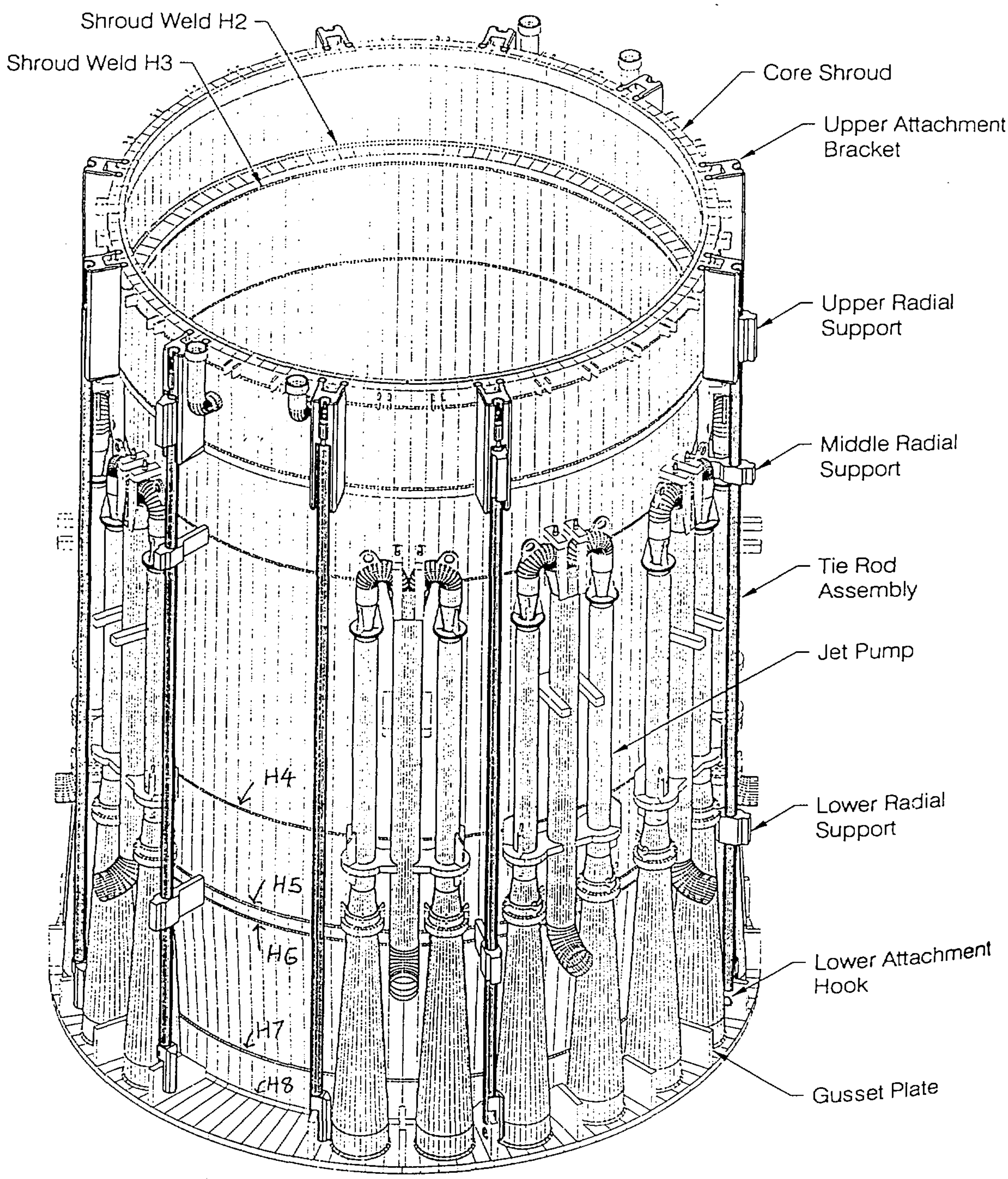


Figure 5-1 BWR core shroud and jet pumps (from McCurdy & Marrone, 1996).

The USNRC staff has developed a probabilistic safety perspective regarding shroud separation at the lower elevation for the above two Mark 1 BWR/3 units. The assessment estimated the potential contribution to core damage frequency due to the cracked shroud. Assuming that a 360°, nearly through-wall shroud crack did exist, a large rupture of either a steam or recirculation line would have to occur to generate sufficiently large loads to move the shroud. Probabilistic risk assessments categorize such ruptures to be of low probability and none have ever actually occurred at an operating nuclear plant. Therefore, the unlikely occurrence of a 360° nearly through-wall crack along with a large pipe break would be necessary to pose any incremental risk. In additional, for welds in the upper portion of the shroud, such extensive degradation should be detected during normal operation (e.g., by power/flow mismatch or noise monitoring due to upper shroud motion). Finally, the shroud may not move in the most adverse manner during these events, and core cooling and reactor shutdown would be achieved with no adverse consequences.

Considering the above evaluations, the USNRC staff has made conservative estimates of the risk contribution from shroud cracking and concluded that it does not pose a high degree of risk at this time. However, although immediate plant shutdowns for inspections are not warranted, degradation of the core shroud does impact plant safety. The core shroud provides the important functions of properly directing coolant flow through the core, maintaining the core geometry, and providing a refloodable volume under postulated accident conditions. The USNRC staff therefore considers that 360° cracking of the shroud is a safety concern for the long term based on: (1) potentially exceeding the ASME Code structural margins if the cracks are sufficiently deep and continue to propagate through the subsequent operating cycle; and (2) elimination of a layer of defense-in-depth for plant safety.

Notwithstanding the capability to evaluate the acceptability of cracked core shrouds for continued operation, USNRC continues to believe that for many of the operating BWRs that have core shroud materials susceptible to stress corrosion cracking, repairs or additional modifications to inhibit cracking will be necessary to assure structural integrity of the shrouds in the long term.

5.1.2 Jet Pump Hold-down Beam Cracking

In 1993, a jet pump hold-down beam ailed by IGSCC at a Mark 3 BWR/6 after about 9 years of service, and two additional hold-down beams had ultrasonic indications of cracks. In addition, numerous jet pump failures had occurred in the late 1970's and early 1980's. An USNRC Bulletin (IEB-80-07, "BWR Jet Pump Assembly Failure") was issued that requested that BWR-3 and BWR-4 licensees perform operability surveillance on the jet pump assemblies. The failure at the BWR/6 was different than prior failures because it occurred in a different location. GE examined the failure and concluded that once a crack begins to grow, it can propagate to failure in less than one operating cycle. Following GE recommendations, the licensee for the BWR/6 replaced all of the jet pump hold-down beams with a new design to decrease the likelihood of failure. Additional GE recommendations to reduce the likelihood of beam failure in BWRs included a reduction in preload and a new heat treatment to reduce the minimum yield strength of the Inconel X750 material. The hold-down beams in service today represent a combination of the above conditions. GE currently recommends that licensees with BWR 4, 5, or 6 design reactors replace their jet pump hold-down beams after 8 years of cumulative power operation.

5.2 PWR Steam Generator Tube Integrity Issues

The thin walled tubing of the steam generator constitutes more than half of the reactor coolant pressure boundary (RCPB). Thus, maintenance of the structural and leakage integrity of the steam generator tubing is a requirement of the 10 CFR Part 50, Appendix A. Nonetheless, steam generator tubing in the U.S. has exhibited widespread degradation by a variety of corrosion and mechanical mechanisms. These problems have cause seven steam generator tube rupture events, numerous forced shutdowns due to steam generator tube leakage, extensive tube repairs and outage extensions, steam generator replacement at eleven plants, and significant personnel exposure.

The primary modes of degradation affecting the steam generator tubing in the U.S. include primary water stress corrosion cracking (PWSCC) and outside diameter stress corrosion cracking (ODSCC) along with wear and fatigue (to a lesser extent). During the past six years, these mechanisms have been the major causes of steam generator tube repair (i.e., plugging or sleeving) and leakage incidents up to and including steam generator tube ruptures.

PWSCC occurs mainly in the expanded portion of the tube within the tubesheet (refer to Figure 5-2), the expansion-transition area, in the U-bends of the smaller radius tubes, and at tube-to-tube support plate intersections which have experienced denting. experience with PWSCC through the mid-1990s involved mainly axially oriented cracks. However, circumferentially oriented PWSCC has become widespread in recent years, particularly at the expansion-transition location in the tube sheet. Circumferentially oriented PWSCC has also been observed in the U-bends of the smaller radius tubes.

ODSCC is frequently associated with generalized intergranular attack (IGA) and usually occurs in stagnant flow regions where impurities from the bulk water can concentrate, such as in the tubesheet area of partial-depth expanded tubes, the sludge pile on top of the tubesheet, and the tube-to-tube support plate intersections. ODSCC, like PWSCC, occurs in both axial and circumferential orientations. Circumferential ODSCC occurs at the expansion-transition locations of tubes and at the tube-to-tube support plate intersections that have experienced significant denting. Axial ODSCC occurs usually at the tube-to-tube support plate intersections. In addition to cracking of unrepaired steam generator tubes, cracking has also been observed in the expansion region of the parent tube at sleeved joints.

Finally, there is recent evidence of cracking of steam generator tube plugs used to plug defection tubes. Degradation of Inconel 600 steam generator tubing continues to result in the need for improved inspection technology, alternate tube repair criteria, and improved primary-to-secondary leak rate monitoring programs. During the last three to four years, the USNRC has focused on the development of alternate repair criteria for specific types of flaws (e.g., ODSCC confined to within the thickness of the tube support plate, PWSCC at the expansion-transition).

5.3 PERFORMANCE OF MOTOR-OPERATED VALVES

Most fluid systems at nuclear power plants depend to a large extent on the successful operation of motor-operated valves (MOVs) in performing their system functions including the passive safety systems of the new design. For example, MOVs may be required to (1) open to

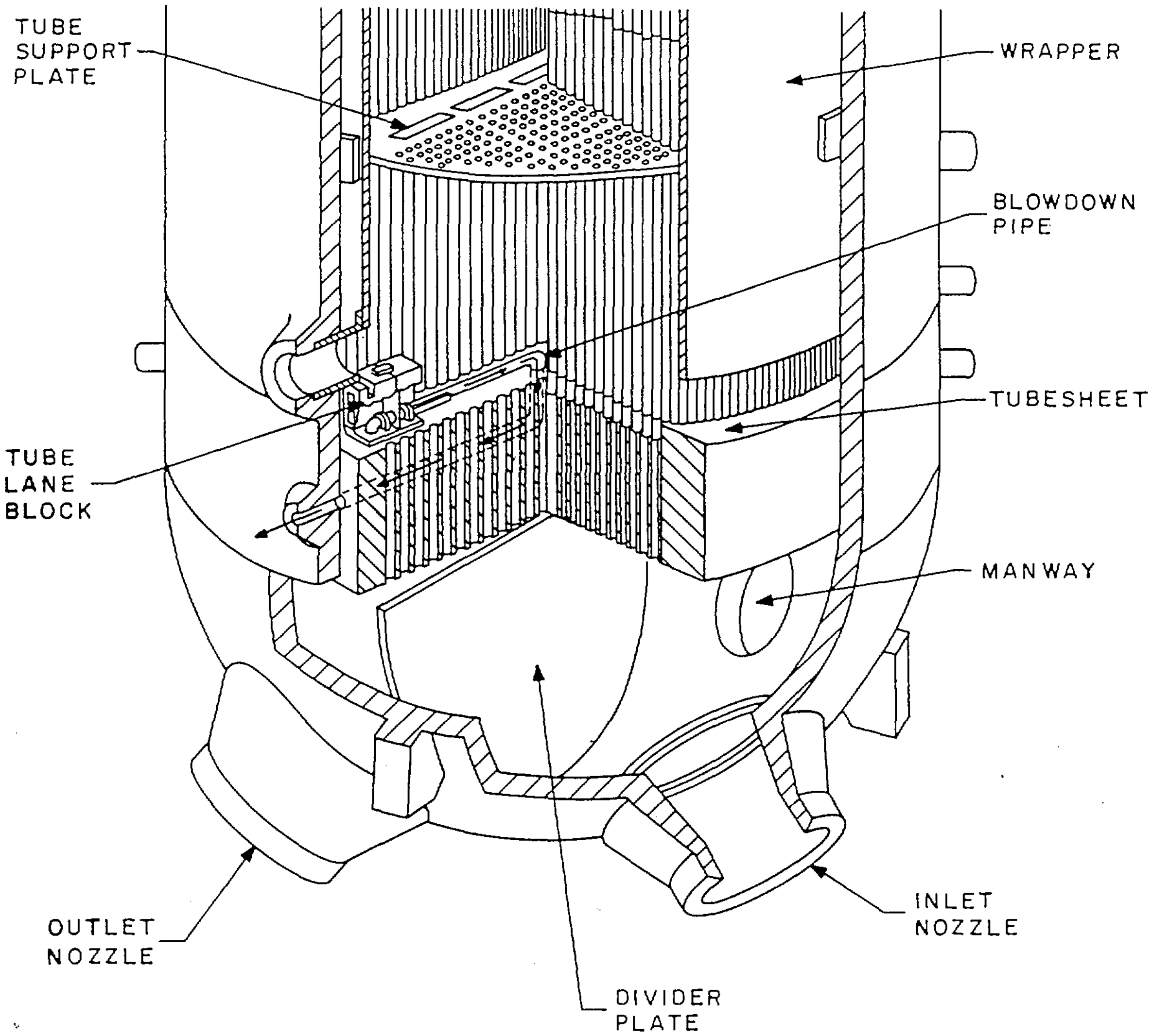


Figure 5-2 Steam Generator Channel Head.

allow cooling water to be provided to the reactor core, steam generators, or containment building; (2) to open to allow steam flow for turbine-driven pumps in safety systems; or (3) to close to prevent loss of coolant from the reactor core or to isolate the reactor containment. Other MOVs may be used to control flow in order to maintain the proper balance of fluids for the production of electric power. To ensure plant safety and to maintain plant availability, MOVs must be capable of performing their functions under design-basis conditions; which may include high fluid differential pressure and flow, high ambient temperature; and degraded motor voltage.

Operating experience at nuclear power plants has revealed weaknesses in many activities associated with MOV performance, including inadequacies in engineering design, shortcomings in maintenance programs such as inadequate procedures and training, and inservice testing under inappropriate test conditions.

The most significant MOV problems in the U.S. result from the weakness in the initial design and qualification of MOVs before their installation in nuclear power plants. The weakness in MOV design and qualification contributed to instances where (1) thrust and torque requirements to operate valves were underestimated as a result of the underprediction of friction, or design-basis differential pressure; (2) motor actuator output was overestimated by failing to determine design-basis minimum voltage, ambient temperature effects on motor output, or load sensitive behavior; (3) structural capability of MOV components was insufficient; and (4) the potential for pressure locking and thermal binding of gate valves was inadequately considered.

Operating experience has revealed weaknesses in procedures for MOV maintenance and training of MOV personnel, in the evaluation of the cause of MOV problems, and in the trending of MOV problems at U.S. nuclear power plants. NUREG-1352, "Action Plans for Motor-Operated Valves and Check Valves," described actions to organize the activities aimed at resolving the concerns about MOV (and check valve) performance. Among those actions were evaluation of the current regulatory requirements and guidance applicable to MOVs, development of guidance for and coordination of USNRC inspections, completion of USNRC MOV research programs, implementation of the research results, and providing MOV information to the nuclear industry.

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