POLICY ISSUE

(NEGATIVE CONSENT)

August 27, 1991 SECY-91-273

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: REVIEW OF VENDORS' TEST PROGRAMS TO SUPPORT THE DESIGN CERTIFICATION OF PASSIVE LIGHT WATER REACTORS

Purpose: To present the staff's recommendations for reviewing, monitoring, and approving the vendors' test programs to support the design certification of advanced, passive light water reactors.

Background: In "Early Review of AP-600 and SBWR Research Needs," SECY-91-057, the staff presented its plan for performing an early review of important new safety features of the two advanced, passive light water reactors (LWRs): the Westinghouse Electric Corporation's AP-600 and the General Electric Company's simplified boiling water reactor (SBWR). The goals of the review are:

(1) Identify those new safety features that will most likely need a lengthy verification program.

NOTE: TO BE MADE PUBLICLY AVAILABLE WHEN THE FINAL SRM IS MADE AVAILABLE

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(2) Identify missing information or possible weaknesses in the vendors' development and verification programs for each feature identified in item 1.

(3) Recommend improvements in the vendors' testing or other verification programs, as needed. Initiate action on confirmatory NRC programs, if needed.

The last of these three goals is important because of the requirements of Section 52.47(b)(2)(i)(A) of Title 10 of the Code of Federal Regulations which mandates that, **in the absence of a prototype, the applicant must have adequately demonstrated the performance of safety features through tests, analyses, or experience; have evaluated the interdependent effects among the safety features of the design and found them acceptable; and have shown that the analytical tools themselves have a sufficient database to establish their ability to perform the requisite safety analyses.**

The advanced, passive reactor designs have a number of unique features that distinguish them from both the current generation of LWRs and the evolutionary ALWRs. All of the safety systems are designed to be passive, relying on natural forces, such as buoyancy and hydrostatic differentials, to drive safety injection flows. However, the vendors have also defined "passive" to include check valves and components that rely on stored electrical, mechanical, and hydraulic energy, such as batteries, springs, and compressed gas, respectively. In addition, both the pressurized water reactor (PWR) and boiling water reactor (BWR) designs include automatic depressurization systems. These designs also include nonsafety grade, active systems as the first line of defense in the event of transients. Several systems are included that in the past have been considered safety-related, such as start-up (auxiliary) feedwater, diesel generators, and pumped residual heat removal systems. In SECY-90-406, "Quarterly Report on Emerging Technical Concerns," the staff identified the role of these nonsafety systems in the passive designs as an emerging technical issue. The unique features of these plants have prompted the staff to identify concerns about system performance in such areas as the long-term reliability of check valves, the performance of automatic depressurization valves, and the thermal-hydraulic interactions between passive safety systems, and, when applicable, between passive and pumped systems.

The staff has focused its efforts on Westinghouse's AP-600 design, about which the staff has substantial, though not complete, information regarding the vendor's planned test and analysis programs. The staff has also recently begun reviewing the SBWR, following a briefing by GE on the current status of the design and research programs. The staff is reviewing issues requiring lengthy or complex verification programs, as described in SECY-91-057. The staff has summarized the results of the review herein. Enclosure 1 provides detailed discussion of areas in which the staff believes that research by the vendor, or NRC, or both will be needed. The staff is assessing the adequacy of the vendor's test and analysis programs for AP-600 to resolve these key issues, and has uncovered a number of weaknesses that the vendor must address by conducting additional testing. Enclosure 2 provides specific details of these deficiencies. The staff has begun to evaluate weaknesses in GE's SBWR test and analysis program.

The staff also believes that the NRC should develop a structured process for reviewing the vendors' test and analysis programs to expedite the review and address the staff's concerns in a timely fashion. In SECY-91-239, "Preapplication Reviews of Advanced LWR Designs," the staff outlined the elements of the preapplication evaluation of the vendors' testing programs. This paper is the first of a series of papers that will deal with the issues discussed in SECY-91-057 and SECY-91-239, setting out the staff's recommendation to develop a structured review process. In a future paper, the staff will discuss the need to conduct large-scale integral systems testing in conjunction with design certification for the advanced, passive light water reactors.

<u>Discussion:</u>

The Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES) reviewed preliminary design information and briefings from the applicants and have identified 14 major types of issues that need early attention:

1. The thermal-hydraulics of reactor and safety systems

2. Containment cooling

3. SBWR stability

4. Systems interactions

5 Analyses of design basis transients and accidents

6. Design basis analytical methods

7. Severe accident assessment

8. Reliability assurance

9. Advanced instrumentation and controls

10. Valve performance

11. Design codes and industry standards

12. Containment performance

13. The reliability of modular construction

14. New materials and aging.

The staff has already initiated programs that address several of these issues. For instance, a RELAP5 thermal-hydraulic model for the AP-600 is currently being developed to permit the staff to evaluate possible accident conditions. A thermal-hydraulic model for the SBWR is being developed. The staff has also initiated programs (1) to address SBWR stability, (2) to develop review criteria for advanced instrumentation and control system software, and (3) to develop methods for assessing the reliability of passive systems and for establishing reliability goals for important passive and active systems. The staff will continue to update the list of research needs and their disposition throughout the early reviews of plant designs and research programs. NRR and RES will coordinate these efforts. The staff expects the list to change as new information becomes available, especially regarding SBWR. The staff will inform the Commission about issues that could significantly affect design certification schedules or raise new policy questions.

The staff has also developed specific recommendations for additional vendor testing to support the AP-600 design certification and is developing similar recommendations for SBWR. To resolve these issues in a timely fashion, with minimal effect on certification schedules, the staff must transmit these recommendations to the vendors. The NRC must also assure itself that the vendors perform test programs to properly address relevant issues.

The staff believes that the NRC should develop a structured process for interacting with the vendors regarding their planned programs for testing and analysis. In the case of the AP-600, the staff has met several times with Westinghouse to discuss the planned experimental programs to support design certification, and has reviewed material supplied by the vendor at these meetings. The staff has discussed aspects of the test programs in which it identified weaknesses that require additional vendor tests. However, the staff has not formally transmitted these concerns to the vendor.

Both Westinghouse and GE are planning to complete a substantial amount of testing before submitting formal applications for design certification. The staff is concerned that the vendors are performing these test programs without sufficient review and oversight by the NRC. If the NRC delays implementing formal review and oversight activities until the formal applications for design certification are submitted next year, it could delay significantly the schedule for design certification because additional testing and reviews of the data may be required. Therefore, the staff will institute a formal relationship with the vendors before receiving the application for design certification. The staff believes this will allow it to expedite the review process and will allow the vendors to address the staff's concerns about test programs with a minimum effect on the design certification schedule. The staff proposes the following formal review procedure:

(1) The NRC will require the vendors to submit their test plans to support design certification before performing any testing. The NRC will assess the adequacy of the planned programs, provide comments on the planned testing identifying weaknesses and deficiencies, and identify additional tests, including possible modifications to test facilities and their instrumentation, that are necessary to correct deficiencies and provide the data needed for design certification.

    Testing that is already planned and is scheduled to start in the immediate future can be conducted as planned prior to NRC review. Full documentation on these tests, including test plans and qualified raw data, should be submitted to the NRC upon completion.

    The staff would require full documentation and test reports from vendors within 120 days for those tests that have already been completed. These reports should include all qualified (reviewed and calibrated) raw data taken from the tests.

(2) After reviewing and approving the planned test program, modified as appropriate, NRC will provide

personnel, to be stationed at the vendor's test sites, to monitor testing activities.

(3) The NRC will require the vendors to submit their qualified raw data and analyses for the NRC staff to review. The NRC will perform audit analyses, as necessary.

(4) The NRC may require the vendors to perform additional tests beyond those originally approved, if information from other tests or analyses indicates that previous testing and analyses are not adequate to satisfy the 10 CFR 52.47 requirements.

(5) The NRC may identify additional confirmatory testing to be done at NRC's expense in the vendor's facilities, beyond the testing required for design certification.

(6) The NRC will select certain tests for pre-test predictions by vendor and/or NRC codes.

Coordination: In the March 21, 1991, SRM on SECY-91-057, the Commission indicated that the staff should obtain clearance from the Office of the General Counsel (OGC) to avoid conflicts of interest in accomplishing proposed NRC tasks. Since each task is unique and will likely require the NRC's attention individually, the staff will seek OGC clearance accordingly. However, OGC has indicated that the NRC may use the vendors' facilities to perform confirmatory testing, as described in item 5 above, upon receiving a waiver from the Executive Director for Operations. The staff expects to seek such waivers for individual cases.

Conclusions: The staff believes that its recommendations regarding these matters conform to the Commission policy on advanced reactors. The staff also believes that the NRC should establish a structured relationship with the vendors to proceed with design certification for the advanced, passive light water reactors with minimal effect on schedules.

Resource Impact: The necessary resources to perform the procedures detailed in this paper are currently budgeted and available; no additional resources beyond those available should be required.

Recommendation: That the Commission

(1) Note the staff's activities to identify research needs for advanced passive reactors, and

(2) Note the staff's positions on reviewing the vendors' test programs as detailed in the discussion above.

The staff will continue to perform these activities as described herein unless instructed otherwise by the Commission within 10 working days.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. List of AP-600 and SBWR Research Needs that Require Early Attention
2. Issues Requiring Vendor Testing for AP-600 Design Certification

SECY NOTE: In the absence of instructions to the contrary, SECY will notify the staff on Thursday, September 12, 1991, that the Commission, by negative consent, assents to the action proposed in this paper.

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ENCLOSURE 1

LIST OF AP-600 AND SBWR RESEARCH NEEDS THAT REQUIRE EARLY ATTENTION

1. Reactor and Safety System Thermal-Hydraulic

Design features proposed for the passive LWRs include the use of passive, gravity-fed water supplies for emergency core cooling and natural circulation safety-grade decay heat removal for the AP-600 and SBWR, and natural circulation cooling within the SBWR core for all conditions. Both plants also employ automatic depressurization systems (ADSs), the operation of which are essential during a range of accidents to allow adequate emergency core coolant injection. The low-flow regimes associated with these designs will involve natural circulation flow paths not typical of current LWRs. Modeling of natural circulation flows will require accurate calculation of small buoyant forces. The analytical codes used for current LWR analyses have not been validated for these new arrangements of piping and system components under the low-flow regimes expected. Additional thermal-hydraulic considerations need to be addressed, such as the operation of gravity-fed injection systems and natural circulation heat exchangers at both high and low pressures, and in the presence of non-condensible gases; two-phase flow in reactor coolant pumps; critical flow in the automatic depressurization valves over the range of expected operating conditions; pressure drop characteristics of loop components, especially check valves; and boron transport under gravity-driven system conditions. Additional thermal-hydraulic test data are needed in these areas, both for model development and improvements for existing codes, and for confirmation of the validity of existing models under low-flow conditions. While the development of needed test data and the validation of thermal-hydraulic codes are largely the responsibility of the reactor vendors, some NRC tests may be warranted to ensure adequacy of the vendor data and to provide sufficient independent data to validate NRC audit tools, such as RELAP5, so that independent safety analyses can be performed by the staff. The construction of test loops to obtain data, performance of requisite tests, and the subsequent data analyses required for code validation all require long lead times. Decisions on what types of test facilities are needed, the scales of those facilities, the scope of testing, and the degree of NRC involvement in the testing need to be made in a timely manner to support design certification activities.

2. Containment Cooling

Cooling for the AP-600 containment involves the use of gravity-fed water on the exterior of the containment shell and natural circulation on both the shell interior and exterior. The scale is large and the design is unlike previous systems. The need for testing of this design is recognized by the vendor, who is planning to perform several series of tests at various scales, including experiments in an approximately 1:8.5 linear scale model. The staff intends to review the Westinghouse plans for these tests, and is investigating possible confirmatory tests beyond those planned by the vendor, to obtain data in such areas as; modeling of natural circulation flow patterns during hydrogen combustion, condensation heat transfer, water film behavior, and drainage behavior back to the in-containment refueling water storage tank. Independent scaling studies may be used to help in evaluating the vendor's test program.

The SBWR passive containment cooling system (PCCS) is also a natural circulation system involving components both inside and outside of the containment. Heat exchangers (HXs) are located in large water pools outside of the containment boundary. The tube side of the HXs are open to the containment atmosphere (drywell). In the event that steam is released into the drywell, as, for example, during ADS operation, the increased pressure in the drywell will help to drive flow through the HXs. Steam is condensed and the condensate flows to the gravity-driven emergency core cooling pools, while non-condensible gases are exhausted to the pressure suppression pool (wetwell). The HX shell-side water pools are allowed to boil as the containment is cooled, with the steam exhausted to the environment. The vendor has completed a 1:400 volumetric scale test of the PCCS, and is planning a full-scale, separate effects test of the heat exchanger, and a 1:20 scale integral system test. The staff will be reviewing the vendor's test program to determine if it is adequate to address key issues, such as heat transfer in the presence of non-condensibles and operation of the non-condensible gas venting system.

3. SBWR Stability

Recirculation flow in the SBWR core region is provided by buoyancy-driven natural circulation, rather than by motor-operated pumps. Core coolant flow may thus be low under some transient conditions, leading to unstable power and flow oscillations. Considerations include differences from current designs, such as reduced core height and the presence of a chimney above the core. An early NRC assessment is needed of the vendor's analytical and experimental basis for demonstrating nuclear/thermal-hydraulic stability, and to identify any tests or analyses that may be needed to support staff technical evaluations of the issue.

4. Systems Interactions

In the design of the current generation of operating reactors, redundancy and independence have been designed into the protection systems so that no single failure results in loss of the protection function. Because the new passive LWR designs incorporate significant changes from the familiar current LWR designs and place a higher reliance on individual systems, a thorough understanding is needed of these designs with respect to systems interactions.

Interactions may occur between the passive safety systems in AP-600 (e.g., the core makeup tanks and accumulators) and SBWR (e.g., ADS system and isolation condensers). New configurations of safety systems are used, such as coupling the AP-600 pressurizer to the core makeup tanks. In addition, the operational philosophy for these plants calls for the first line of defense in the event of a transient to be the non-safety active systems; this may lead to interactions between the active systems and the passive systems that could be detrimental to the operation of both. In addition, there is a close coupling in both plant designs between the reactor coolant system and the containment during an accident. The staff must assess the design features of the passive plants, to determine if planned testing is adequate to evaluate potential systems interactions, and to ensure that deleterious systems interactions are either eliminated or are extremely limited.

The staff is also considering performing other independent systems analyses to look for potential multiple or consequential failures. The purpose of these analyses is to evaluate overall behavior in various reactor systems and in instrumentation and control systems to identify:

(1) significant failure modes and accident scenarios;

(2) important components and systems needed for safe operation;

(3) minimum instrumentation requirements needed to monitor accident behavior; and

(4) vulnerabilities under degraded plant conditions conducive to common-mode failures, such as earthquakes, fires, and floods.

Methods to provide in-service inspection, monitoring, and diagnosis of degradation and I&C performance also need to be evaluated for adverse systems interactions, and acceptable methods identified therefor.

5. Design Basis Transient and Accident Analysis

An applicant's demonstration of the safety of a nuclear power plant includes the analysis of plant response to postulated disturbances in system performance parameters, and to postulated malfunctions or failures of equipment. These safety analyses are one of the main focuses of the Commission's licensing review. Chapter 15 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," identifies the specific analyses required for staff review of current LWR license applications. The unique design features of the passive reactors have the potential to introduce accident and transient behavior that is significantly different from the current designs. It is likely that revisions to current transient and accident acceptance criteria, as well as to analysis assumptions, will be required. The staff must determine the scope and content of such revisions.

6. Design-Basis Analytical Methods

As discussed in the previous section, the set of postulated events that will form the design basis for the proposed AP-600 and SBWR designs may be different from that of current LWR designs. Accordingly, an assessment needs to be performed of the capability of current vendor and NRC analytical tools to model appropriately these designs, given the new and unique system designs. Discussions will be held with the vendors to identify any changes required in vendor codes to assure that transients and accidents are properly modeled. The staff also needs to develop independent audit capability to evaluate unique aspects of the AP-600 and SBWR designs.

7. Severe Accident Assessment

Unique design features of the advanced passive reactors could produce system and containment performance that is markedly different from current generation LWRs under severe accident conditions. The staff needs to assess those phenomena associated with the progression of severe accidents, including events that can lead to challenges to the passive containment designs. From this, the staff can then determine the extent to which the current experimental data base and analytical models are adequate to support assessment of the design, and which areas will require additional experimental data and analytical model improvement to evaluate the severe accident behavior of passive reactor designs. Important issues in this area include debris coolability; hydrogen generation and control; and containment performance under severe accident thermal and mechanical loads.

The staff will evaluate the results of the industry-sponsored ACE/MACE debris coolability tests to determine if further data are needed to support the proposed surface area criterion of 0.02 m^2/MWt, in light of the larger mass of core material in the passive ALWRs per unit of thermal power, which would result in deeper debris beds than predicted for current generation plants. The vendors' containment testing programs will be evaluated to determine if additional data will be required to assess hydrogen and containment performance issues; the degree of NRC involvement, if any, in additional testing must also be determined.

8. Reliability Determination and Assurance

Simpler, passive systems are used in the passive designs to replace various multiple active systems. The applicants and the NRC need to develop a program for assessing and assuring the reliability of the new designs. Several related questions arise concerning risk assessment for these passive systems and the broader question of defense-in-depth. This would include the development of a framework for reliability assurance starting with the initial design phase and continuing through plant operation.

(a) How is failure defined for a passive system? Precursors to possible failure modes for passive systems (e.g., flow blockages, tank or vessel failures, changes in heat transfer characteristics, embrittlement and fatigue cracks in components) may be different from precursors to failure modes for active systems. Failure modes for passive systems may be difficult to identify and to model in a risk assessment. An assessment also needs to be made as to whether check valves in passive designs should be considered active components and postulated to undergo active failures.

(b) How does one quantify failure rates for passive systems? Central estimates of the failure rates may be very low, but have large uncertainties because of the lack of data. If there were sufficient redundancy and diversity in the means of accomplishing the safety function, then these uncertainties may not be important.

What reliance can be placed on very low postulated failure rates (e.g., in the range of 10^{-5} to 10^{-6}), given the level of redundancy and diversity currently employed in the passive reactor designs, and when are such small rates not verifiable? Is there some recognizable limit in failure rate below which values should not be accepted in a risk assessment? Also, degradation mechanisms need to be identified and their acceptable propagation rates need to be established.

(c) Do passive systems adequately support the defense-in-depth concept? If not, should there be any special requirements for mechanical design, surveillance, in-service testing, inspection, and maintenance for the passive systems?

9. Advanced Instrumentation and Controls

Instrumentation and control (I&C) systems in advanced reactors will make extensive use of equipment and design practices that are expected to be significantly and functionally different from current designs. These designs will involve, but not be limited to, the use of microprocessors, digital systems and displays, fiber-optics, multiplexing, and the use of different isolation devices to achieve the needed independence and redundancy. The use of new control and display systems and the potential broad use of artificial intelligence and expert diagnostic systems with optional reliance on manual operations will have a major impact on the role of the operators and their performance during the range of normal and postulated accident conditions. Industry standards need revision for the qualification of advanced I&C components and software to ensure operability in the nuclear plant environment during normal operation, accidents, and post-accident conditions. Research and standards development related to digital systems and software are needed to support I&C system reviews for these plants.

A number of separate considerations regarding the use of advanced I&C for monitoring and diagnostic purposes in the passive plants, and the impact on the role of the operator need early attention. These include:

Qualification and Development of Acceptance Criteria

(a) Development of methods for evaluating the quality of the software used in digital safety and control systems. These include considerations for software reliability and the development of acceptance criteria for areas such as information management systems, software languages, software testing, and expert/artificial intelligence systems.

(b) Support is also needed for the development of Standards and Guides to ensure operability and environmental qualification of equipment. Technical data must be developed to understand the effects of plant environment on equipment operability and long-term aging. Confirmatory tests and acceptance criteria will be needed to establish significant degradation mechanisms, failure modes, and fragility levels for selected components. Also, methods and parameters useful for monitoring and trending performance over long periods need to be developed.

(c) Specific research will be required to develop and implement an electrical isolation device testing program that examines the effects of fault voltages and currents.

(d) Confirmatory research is needed to evaluate the effects of electromagnetic (EM) and radio frequency (RF) interference and electrical surges on digital systems. Technical bases and acceptance criteria oriented toward hardware concerns with EM and RF issues in digital electronic systems must be developed.

Human Factors

(e) The staff needs to evaluate each vendor's proposed, updated human reliability analysis (HRA) models and data, and should also consider the need to perform independent HRAs of operator performance under postulated accident conditions.

(f) Methods and criteria need to be developed for evaluating interactive computerized procedures and advanced, computer-driven annunciator/alarm systems that may include alarm prioritization and filtering.

(g) Decisions will be needed on the number of control stations needed for multiple unit plant sites, and the need for control room prototypes for validation and verification of both the separate and integrated man-machine systems.

(h) Research is underway and should continue on methods to assess the impact of cognitive omissions and overload issues.

(i) Research has been underway to establish a performance baseline on the man-machine interface against which to assess objectively the degree of improvement offered by advanced designs; this work should continue.

10. Valve Performance

The AP-600 and SBWR reactor safety systems will utilize various check, solenoid, and power-operated valves. Valve performance has been shown to be a major risk contributor for current LWRs. The application of valves in the passive designs may subject the valves to operating conditions different from those routinely encountered in current LWRs, particularly the lower differential pressures present in natural circulation and gravity-fed cooling systems. A determination is also needed as to whether additional work must be done to develop valve specifications and review guidelines for the check valve designs and the SBWR explosive depressurization (squib) valves. Research is also needed to determine the degree of modification needed for ASME standards and NRC regulatory documents on design, fabrication, qualification, in-service testing, and maintenance.

11. Revisions to Design Codes and Industry Standards

A number of design codes and industry standards dealing with new plant construction have been developed or modified recently. However, their acceptability to the NRC has not yet been established. The following are examples.

(a) ASME Section III, Div. 1, Rules for Construction of Nuclear Power Plant Components.

(b) ASME Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components

(c) ACI 349-85, Code Requirement for Nuclear Safety Related Concrete Structures.

(d) ASCE 4-86, Seismic Analysis of Safety Related Nuclear Structures.

(e) ANSI/AISC N690-1984, Design, Fabrication, and Erection of Steel Safety Related Structures at Nuclear Power Plants.

Codes and standards that will have a significant impact on the review of passive reactor designs need early attention relative to accepted NRC practices and guidelines, so that they can be properly used by the vendors in preparing their applications for design certification.

12. Containment Performance

Containments for advanced reactors will be evaluated for their capability to withstand potential challenges from severe accidents. Containment performance criteria for advanced reactors have been considered by the staff. The staff's proposed containment performance criteria in SECY-90-016 were approved by the Commission for the evolutionary LWR designs. Further definition of these criteria has recently been proposed by the ACRS. In addition, the vendors and industry have also proposed criteria in the EPRI ALWR Requirements Document which are intended to implement the Commission guidance provided in SECY-90-016. All of these various efforts are currently under staff review. Best estimate methodology and acceptance criteria must be developed to assure adequate containment performance under severe accident conditions.

13. Reliability of Modular Construction

Another central feature of these passive designs is the use of modular construction. Modular construction has been used in the U.S. for nuclear power plants to only a limited degree; however, it has been used more widely in Japan. Larger and new types of modules are being proposed for safety-related structures in the advanced, passive LWRs, such as in shear wall assemblies.

Technical issues that will be raised relate to the strength and ductility of joints and connections, as well as the appropriate damping for seismic events. Degradation of stiffness during earthquakes is also an issue for steel and concrete sandwich-type shear wall modules. Requirements for QA and QC during transportation and installation, especially interfacing requirements, need attention, as well.

Because design rules for some modules must be formulated based on test data, and because this modular construction method is so central to these designs, research to resolve these issues should be undertaken early.

14. New Materials and Aging

Advanced reactors are being designed with longer service lives, alternate materials, and different design conditions. The NRC needs a comprehensive evaluation of the materials and associated operating environments for the AP-600 and SBWR. Among the new materials applications are low-cobalt or cobalt-free materials for wear applications, and SA508 Class 3 steel for reactor pressure vessels. Additional research is needed to extend current aging and license renewal activities to help identify and characterize potential degradation and aging mechanisms, mitigation practices, replacement plans, surveillance practices, and design practices, to emphasize inspectability and testability of materials and components. Aging impact on newly developed components, such as the SBWR ADS valves, also need to be assessed since no operational performance history is available. Based on the results of these preliminary evaluations, supplementary research will be initiated as necessary to ensure that an adequate technical basis is provided for the design certification reviews.

ENCLOSURE 2

ISSUES REQUIRING VENDOR TESTING FOR AP-600 DESIGN CERTIFICATION

I. General Description of Test Program

The staff has completed its initial assessment of the AP-600 vendor planned testing program. The planned testing program in support of AP-600 design certification consists of a series of separate effects experiments, which are intended to test the performance of individual components or systems, and one small-scale integral test, which is designed to examine systems interactions over a very limited range of conditions. Following is a brief description of the components and systems to be tested, the planned tests, and a discussion of issues identified by the staff which are not addressed in those tests. Staff recommendations for additional tests are also presented. A similar assessment is currently ongoing for SBWR, which will be reported to the Commission in a future SECY paper.

II. Separate Effects Tests

A. Core Makeup Tank Performance

The two Core Makeup Tanks (CMTs) provide high-pressure, safety-grade coolant makeup to the primary system (see Figs. 1 and 2). Each tank contains approximately 2000 ft^3 of cold, borated water. The outlet of the tank, at its bottom, is connected to the direct vessel (downcomer) safety injection line, and is isolated from the primary system by both DC-operated valves and check valves. The top of the tank is connected to both the pressurizer and the primary system cold leg for pressure balancing capability. The line to the pressurizer is always open, with backflow toward the pressurizer prevented by two check valves in series. The lines to the cold leg are isolated by DC-operated valves. In the event of a small break from which the coolant loss exceeds the capacity of the non-safety grade makeup, coolant inventory loss will reduce the pressurizer liquid level. The outlet valves of the CMTs open on receipt of a "low-low" pressurizer signal, and water flows from the CMTs to the primary system under a "manometer" effect driven by the liquid level and density differences between the pressurizer and the CMTs. The CMTs are also tied to the automatic depressurization system (ADS); each stage of the ADS fires in response to successively lower CMT levels.

The safety significance of this system is that it is the only safety-grade source of high-pressure coolant makeup. The connection between the level in these tanks and the ADS is also very important.

The testing proposed by the vendor employs a tank simulating the pressurizer, connected by a line representing the pressurizer/CMT pressure balancing line to a second tank simulating the CMT. A line from the CMT outlet back to the pressurizer tank completes the loop. The drain-down behavior of the CMT will be studied, under the influence of the level differences as described above. Since steam from the pressurizer flows into the cold CMT tank as it drains, condensation behavior in the CMT tank will also be examined.

AP600 - PASSIVE SAFETY SYSTEMS

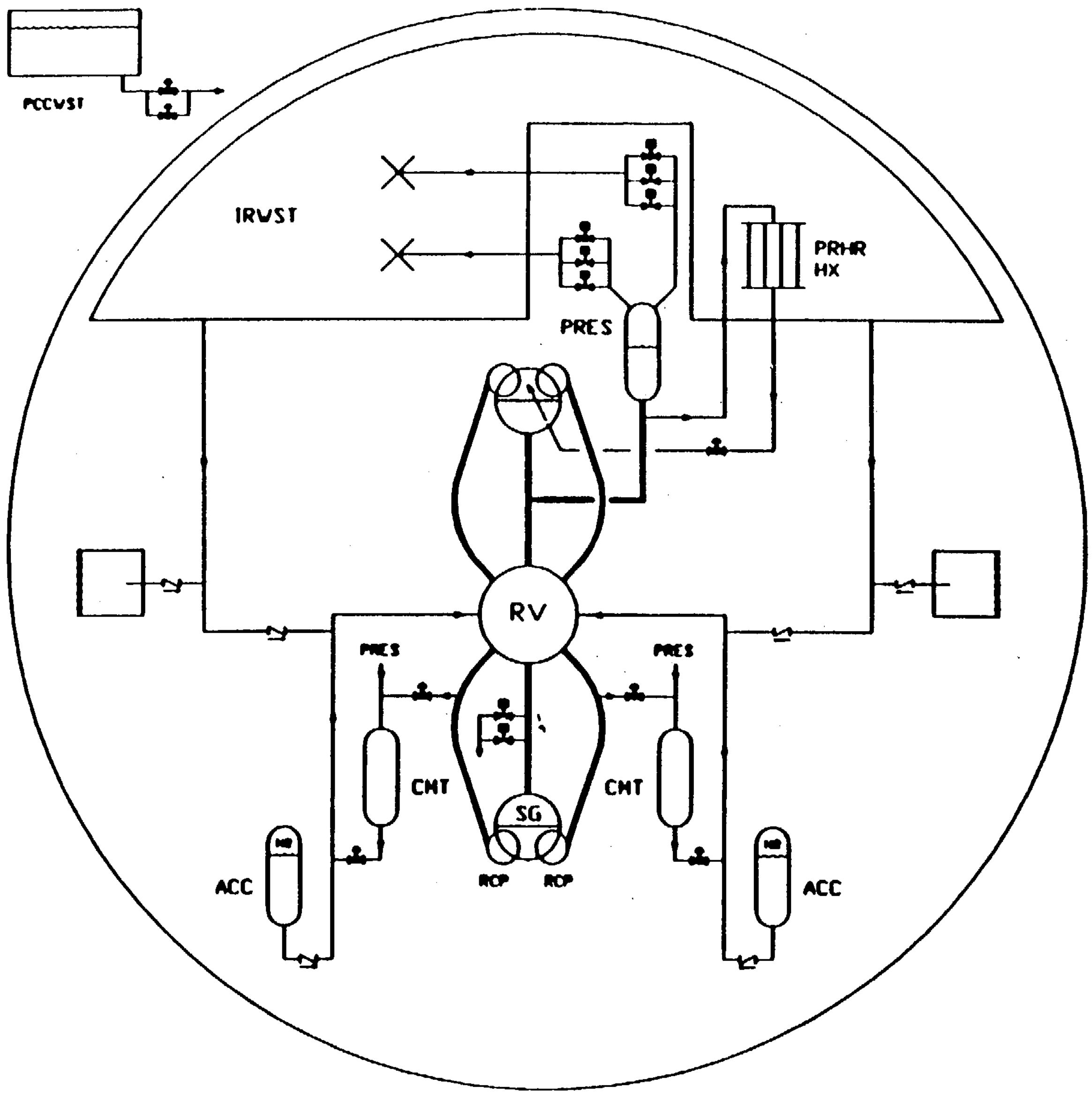

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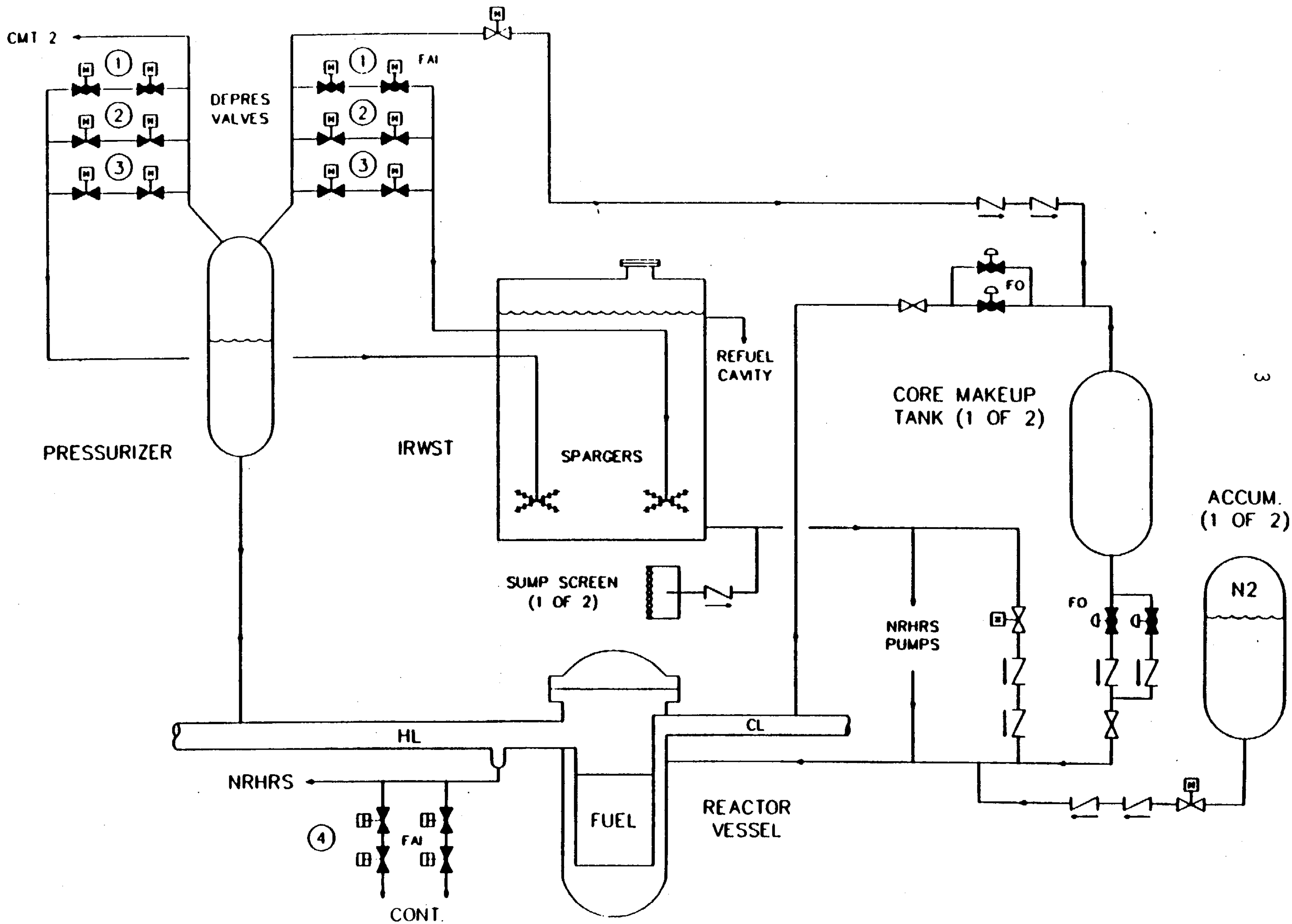
Figure 1

'gure 2

AP600 – PASSIVE CORE COOLING SYSTEM



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The deficiencies identified in these tests are:

a. There is no simulation of the pressure balancing line connected to the primary system cold leg. System interactions involving this line cannot be studied.

b. There is little or no capability to simulate the CMT and the rest of the primary system, especially the impact of pressure oscillations due to steam condensation on injection performance.

c. There is no simulation of ADS behavior resulting from CMT drain-down, or of the effect of system depressurization on CMT operation.

The staff recommends that the test facility be redesigned to include the pressure balancing line from the cold leg, and that tests be included to study interactions between the components in the system. Additional tests that would examine interactions with other safety or non-safety systems could only be accomplished with an integral test facility.

B. Automatic Depressurization System Performance

The ADS, also shown in Figs. 1 and 2, consists of four stages of valves, designed to reduce the reactor system pressure to ambient (containment). All four stages require DC power to operate. The first three stages are connected to a "tree" extending from the upper head of the pressurizer, and exhaust through spargers into the in-containment refueling water storage tank (IRWST), a 400,000-gallon cold water reservoir. The fourth stage is connected to a line coming off of the primary system hot leg, and exhausts to containment atmosphere.

The safety significance of this system is the fact that it <u>must</u> operate to allow the major passive safety injection systems to operate properly. The final stage of safety injection consists of draining the IRWST into the reactor coolant system (RCS). The IRWST inventory is kept out of the RCS as long as the RCS pressure exceeds the hydrostatic head of the IRWST tank, which is about 10 psi. Operation of the final stage of the ADS is necessary to reduce the RCS pressure sufficiently for the IRWST hydrostatic head to exceed that of the RCS and permit safety injection.

Testing planned by the vendor is to be conducted at a facility at ENEA in Italy. Blowdowns of full-scale valves representing the first three stages of the ADS are planned, with the exhaust passing through a full-scale sparger into a quench tank that simulates the IRWST. Depressurization behavior will be examined, and condensation loads on the quench tank will also be measured.

Deficiencies identified in these tests include:

a. No testing of the fourth stage of ADS is planned, including blowdown to a simulated containment environment. Data on critical flow under these conditions and the transition from critical to non-critical flow are needed to permit modeling of ADS behavior.

b. Tests are planned only with steam and saturated water at the inlet of the valves. No tests are planned with a two-phase mixture at the valve inlet. The fluid conditions at the valve inlet affect its critical flow behavior, and thus the depressurization rate of the system.

c. The Passive Residual Heat Removal Exchanger (PRHR HX, described in detail later) is also located in the IRWST. No direct testing of the impact of condensation loads during ADS operation on the PRHR HX is included.

The staff recommends that the vendor's test program be modified to include tests that address the above deficiencies, including simulation of the PRHR HX structure in the quench tank.

C. Check Valve Performance

Check valves are employed in almost all of the passive safety systems to help to provide isolation of the systems when they are not in operation, and to prevent reverse flow when the systems are operating. Several of these valves must change state (open or close) and remain in that new state in response to very small differential pressures, as low as 0.5 psi. The check valves must also operate reliably over long periods of exposure to reactor coolant.

The reliable operation of these valves is a critical safety issue. Failure of valves to open on demand or to stay open would introduce substantial additional flow resistance in natural circulation loops and could seriously degrade flows.

Two sets of tests are proposed by the vendor for these valves. The opening and closing behavior of the valves, and their flow resistances, will be studied at low differential pressures. In addition, a long-term reliability test will be conducted by exposing a valve to postulated "worst-case" reactor coolant conditions for an extended period of time, and examining the valve performance characteristics to measure any changes that might result from such exposure. The latter test will also involve testing of techniques for long-term surveillance of the valve, to ensure acceptable performance in the plant.

The major deficiencies identified by the staff in this program are:

a. No testing is planned to address the dynamic behavior of the valves. Pressure oscillations, such as might occur during ADS operation, could cause the valves to "flutter" or bang against the stop or the seat, damaging the valve or degrading its performance.

b. The long-term reliability tests may be insufficient to establish reliable valve operation over the proposed 60-year lifetime of the plant.

The staff recommends that the test program include the dynamic valve performance tests. Additional information from the vendor is needed to assess fully the adequacy of the long-term reliability test. Methods for accelerating the aging process in the long-term tests need to be examined.

D. Passive Residual Heat Removal Heat Exchanger Performance

The PRHR HX is a safety-grade, full-reactor-pressure system to remove decay heat from the reactor (see Fig. 3). The inlet to the HX comes off of the pressurizer surge line. The HX consists of three banks of 24-ft-long tubes, 450 tubes per bank, submerged in the IRWST. The flow from the pressurizer passes vertically downward through the HX tubes under natural circulation, and returns to the primary system via the steam generator cold side channel head. The HX is isolated from the primary system by DC-operated valves. These valves open either on loss of both main feedwater and start-up feedwater, or on activation of the first stage of the automatic depressurization system. In the event of PRHR system operation, sufficient heat capacity exists in the IRWST to remove decay heat for approximately two hours before the IRWST reaches saturation temperature. Thereafter, as the IRWST boils, steam is released to the containment atmosphere, where it is condensed by the passive containment cooling system (described below) and returned by a system of gutters to the IRWST.

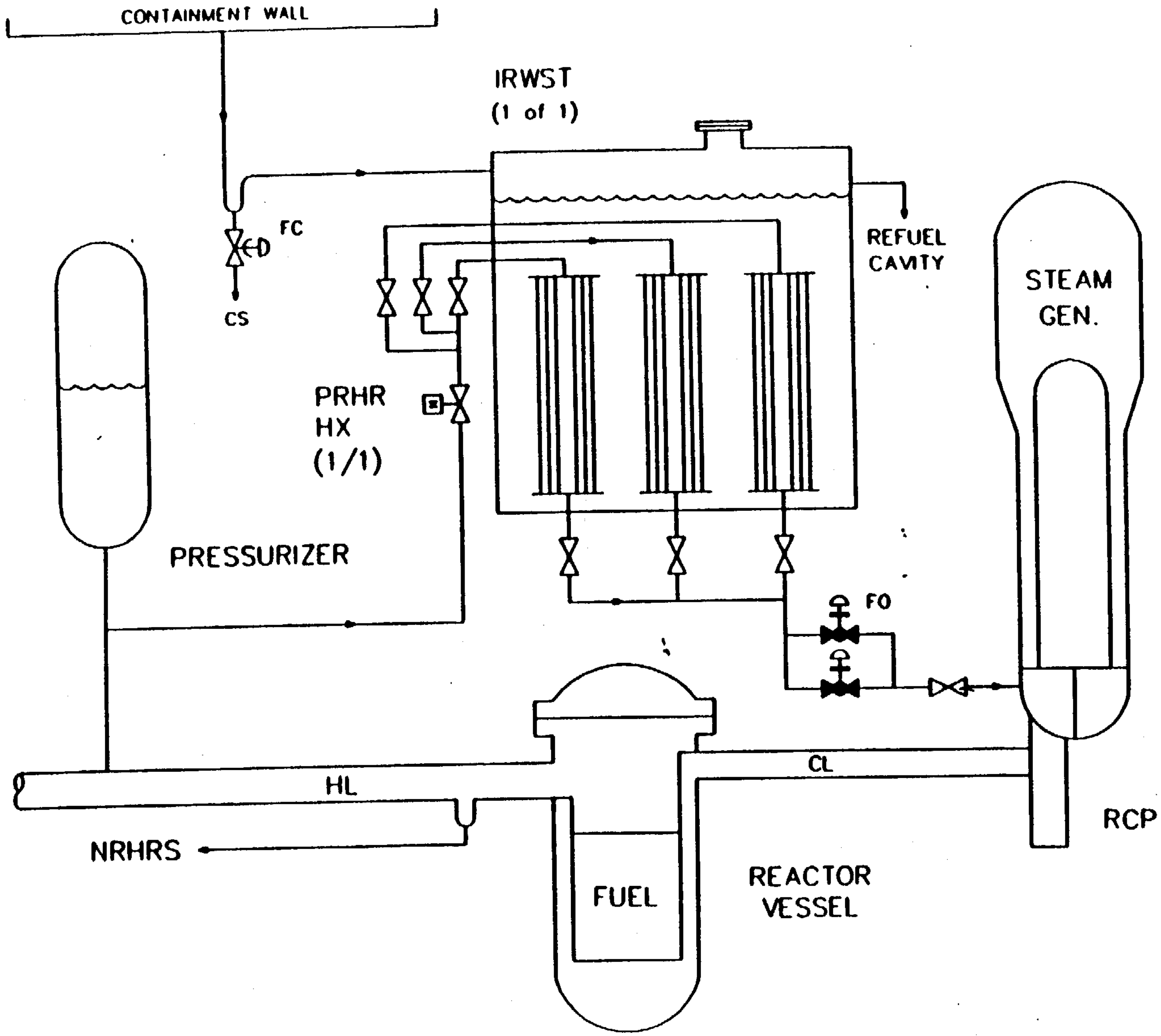
The safety significance of the PRHR system is that it is the only safety-grade, full-reactor-pressure means of removing decay heat in the event of a complete loss of feedwater to the steam generators. A feed-and-bleed process may also be used for decay heat removal, but requires primary system depressurization.

The vendor testing program for this system has already been completed. The tests involved a bank of three full-length heat exchanger tubes, with water under reactor coolant conditions transferring heat to a large tank of cold water. The objective of the test was to measure the heat transfer performance of the tubes. It was found that, instead of the outer surface of the tubes operating in the nucleate boiling regime along their entire length, boiling was suppressed toward the bottom of the tank, due to the increase in saturation temperature with increasing water depth. Heat transfer was therefore substantially lower than predicted. As a result of these tests, the correlation used to predict heat exchanger performance was changed, and the number of tubes was increased to account for the reduced heat transfer per tube.

The staff has not identified specific deficiencies in the tests for this system. However, as noted above, the presence of the PRHR HX in the IRWST does raise concerns about condensation loads during ADS operation and the effect of these loads on HX performance. In addition, the actuation of the PRHR system concurrent with the first stage of ADS opens another flow path for fluid in the pressurizer, in addition to the pressure balancing line to the CMT and the ADS valves themselves. The staff believes that these interactions with other systems should be considered in the scope of other tests, such as the ADS tests described previously, or in integral systems tests, to examine thermal-hydraulic interactions with other systems.

Figure ⌄

AP600 - PASSIVE RHR HX



E. Passive Containment Cooling System Performance

The Passive Containment Cooling System (PCCS) serves as the method by which primary system energy is transported to the ultimate heat sink. It is shown in Figs. 1 and 4. The AP-600 containment is a steel shell, approximately 1.625" thick. As described above, operation of the passive decay heat removal system for an extended period after a transient or an accident will release steam to the containment atmosphere, increasing containment pressure. When the containment pressure reaches a sufficiently high value, the PCCS is actuated. A large reservoir of water is maintained in tanks sitting above the outside of the containment shell. A shield building and baffle are arranged to create a "chimney" effect around the shell. When the system is activated, DC-operated valves open to allow a gravity-fed flow of water onto the outside of the containment. At the same time, air flows under natural circulation, downward between the baffle and the shield building, then upward between the baffle and the containment, cooling the shell by evaporation and convection. The water supply is sufficient to continue for approximately three days, after which time the containment heat load is postulated to be low enough for the energy to be removed by natural convection of dry air alone. The steam being condensed inside the containment flows under gravity along the shell or "rains" back to the operating deck. Most of this water is expected to be collected in a system of gutters that will channel it back to the IRWST, forming a closed cooling loop. Some of the condensate is expected to miss the gutters and to wind up in the containment sump. The water in the sump can also circulate back to the primary system through the downcomer injection line, if RCS pressure is low enough. Check valves provide isolation between the sump and the primary system.

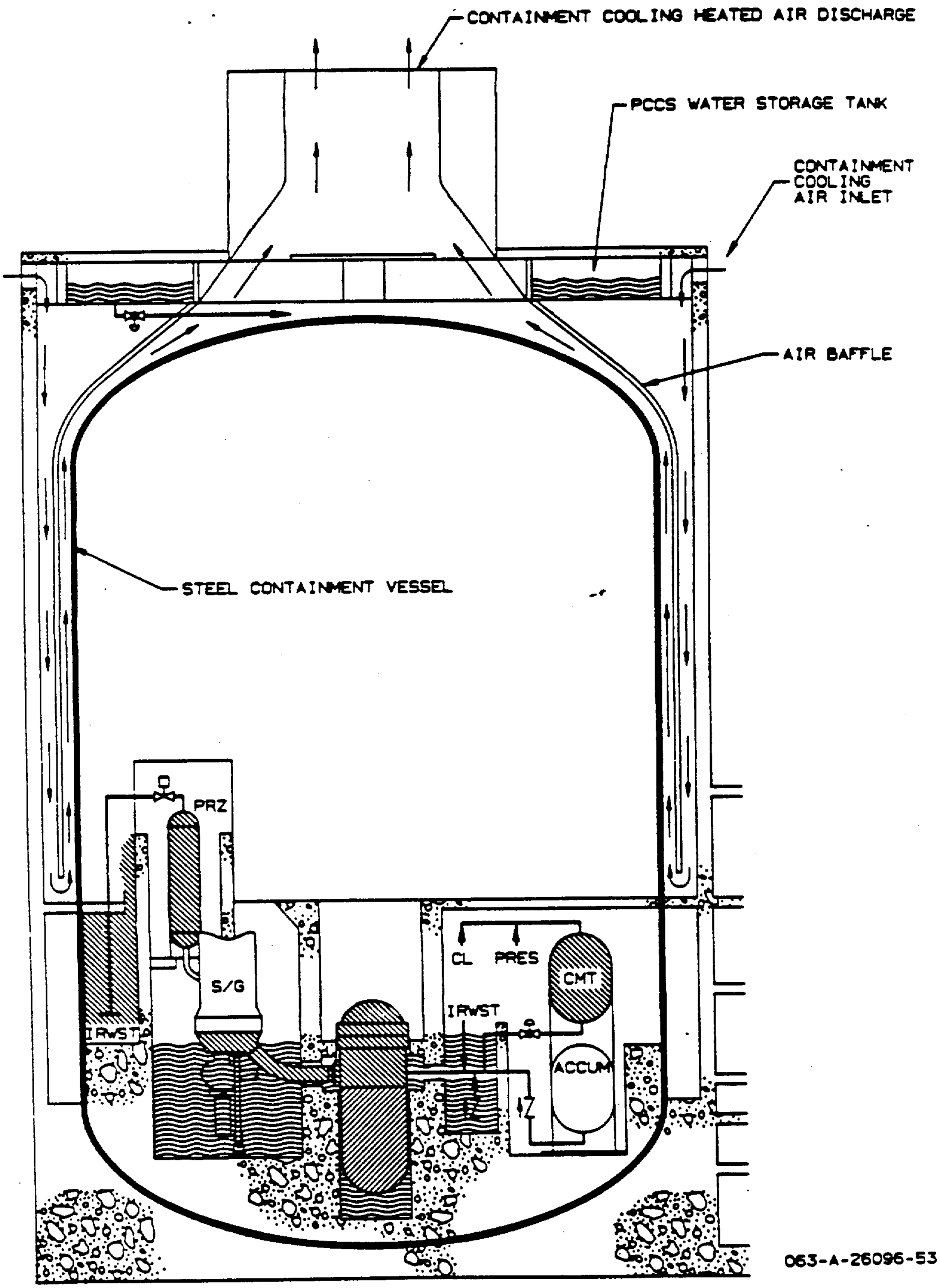
This system is the only safety-grade means of transporting energy to the ultimate heat sink. It is required to maintain the containment at a pressure low enough to prevent failure, and to recirculate coolant back to the RCS.

The proposed vendor test program for the PCCS includes a series of separate effects tests at various scales to examine the heat transfer behavior on the interior of the containment, heat transfer on the containment exterior, and water distribution on the containment exterior. Simple geometry tests have been completed, and those in a more complex geometry are under way. A relatively large-scale facility, approximately 1/9 scale in height and 1/8.5 in diameter, is being constructed for tests of the entire PCCS. The focus of all of these tests is heat transfer behavior. Tests on a full scale angular sector of the containment shell will also be conducted to study water distribution on the containment exterior.

Limitations identified by the staff in this test program include:

a. The reactor coolant system in the large-scale facility has not been simulated. A mock-up of a steam generator will be included to look at the effect of one large structure on fluid distribution. The significance of this omission will be evaluated when test data becomes available.

AP600 PASSIVE SAFETY INJECTION
LOCA LONG TERM



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b. No systematic study has been performed to evaluate scaling of these test facilities.

c. No specific tests are planned to investigate the performance of the PCCS for conditions beyond design basis events. Following a postulated core melt event, containment pressure and temperature conditions may be more severe due to clad oxidation and ex-vessel core/concrete interactions. Greater amounts of non-condensible gases may also be present. The design should be demonstrated to have sufficient robustness to be able to accommodate all credible severe accident conditions. Demonstration of this capability may include a combination of test data with appropriate analyses. The selection of the specific approach with its justification will be the responsibility of the vendor. Therefore, it is premature at this time to decide whether additional testing will be needed until the approach is documented and reviewed.

F. Boron Transport in the RCS

The cold water in the core makeup tanks is highly borated. In the event of an unscrammed transient accompanied by a loss of inventory, the borated CMT water is the means for shutting down the reactor. However, the CMTs inject water at very slow rates, and there is a concern that the boron in the water may be transported through the downcomer and into the RCS in such a way that its distribution may prevent complete effectiveness.

This system provides the only diverse means for reactor shutdown in the event that control rods do not insert.

No vendor tests are currently planned to examine this issue. This is viewed by the staff as a significant deficiency. The staff therefore recommends that tests be developed to study boron diffusion and transport under CMT injection conditions. It may be possible to modify other planned tests in existing or planned facilities to accommodate this need.

III. Integral Tests

A. Long Term Cooling

In the event of a loss-of-coolant accident, either large break or small break, the final state of the reactor coolant system should be at or near containment pressure. If the break is large, the system will depressurize through the break itself. The gas-charged accumulators and CMTs will provide emergency core cooling (ECC) injection early in the transient. The ADS system will also be actuated by the drop in CMT level, but may not have a significant effect on system depressurization behavior. If the break is small, CMT injection will be actuated first, followed by ADS operation; when the system drops below the accumulator gas pressure (assumed to be 700 psia at this time), the accumulators will inject. The system should continue to depressurize as successive stages of ADS are fired, dependent upon the rate of decrease in CMT level. In both scenarios, reduction of RCS pressure to approximately containment pressure should result in ECC injection from the IRWST, flooding the RCS above the level of the loop piping. The system should then enter a long-term recirculatory cooling mode, with coolant boiling off under decay heat, escaping to the containment atmosphere, condensing by action of the PCCS, and returning from the containment shell to either the containment sump or the IRWST to be recirculated to the core (see Fig. 4).

The system behavior described above represents the integrated response of the passive safety systems to an accident. It comprises some of the critical design bases for safety system operation. It is therefore of major safety significance.

Proposed vendor testing in this area covers only the low-pressure segment of the scenarios described, and is oriented primarily toward study of the long-term, stable, recirculatory cooling mode established at the end of the accidents. A small-scale test facility is currently being designed, and is planned for construction at Oregon State University. The scale of the facility is still under study; preliminary plans call for a 1/4 linear (height) scale, and a volumetric scale of 1/226. The facility will represent an integral test bed for system response, including most, but not all, of the passive safety systems that are associated with the RCS. The passive containment cooling system will not be simulated. Major components in the facility will be constructed of quartz or glass to allow flow visualization studies, thus limiting system operation to a maximum pressure of about 65 psia. The primary system, steam generators, reactor coolant pumps, and associated loop and ECC piping will be simulated, as well as the CMTs, accumulators, ADS, and IRWST. The PRHR HX is <u>not</u> included in the simulated IRWST. A blowdown tank will simulate the containment sump. The test program for the facility is not yet available for review.

The staff has identified two major deficiencies in the facility as planned:

a. The lack of a PRHR HX simulation in the IRWST is of concern. Since this component is actuated upon opening of the first stage ADS valves, it becomes an alternate flow path for fluid passing into the pressurizer. The impact of this flow path during the final stages of reactor blowdown and long-term recirculation is not clear.

b. No means exists to simulate the effect of containment backpressure on safety system response. The interplay between containment pressure and safety system operation has been described above in the discussion on IRWST injection. The generation of steam when the RCS is at or near containment pressure could serve to pressurize the system if that steam cannot be relieved quickly. Only a small increase in pressure is required to inhibit injection from the IRWST because of the low hydrostatic head of the tank over the RCS. Interference with IRWST injection could in turn result in additional steam generation, and ultimately in core uncovery, with the potential for serious damage if such a condition should persist.

The staff recommends that the facility design be modified to include a simulated PRHR HX in the IRWST. The staff further believes that the containment backpressure can be simulated by a tank connected to the RCS, in which the pressure can be varied by means of heaters and sprays. Although this would not provide a full simulation of the containment feedback on safety system performance, it would allow sensitivity studies to be conducted over a range of simulated containment backpressures. For example, the driving potential between the IRWST and the RCS can be reduced when there is significant condensation in the containment. Adequate injection flow under such adverse conditions should be verified.

A further staff concern relates to the complete lack of any integral testing to study systems interactions during the high-pressure phase of transients. The behavior of interest includes interactions between the various passive safety systems, as a result of the interconnections, both in the physical sense and in the context of controls, between them. For instance, pressurizer level drop actuates CMT injection; CMT level drop actuates ADS, which affects pressurizer level, which may feed back to CMT behavior, and so forth. In addition, the front-line, non-safety-grade active systems in AP-600 are stated by the vendor to be the first line of defense in the event of a transient, and that they are there, in part, to help prevent challenges to the passive safety systems. The staff is therefore concerned that interactions might occur between the passive safety systems and the active, non-safety systems, in the event that both were activated to deal with a transient. In the staff's view, these concerns can best be addressed through testing in a large-scale, high-pressure, integral test facility. A separate Commission paper will address this issue in detail.

IV. Severe Accident Performance Tests

In the event of a severe, core-melt accident, both safety-grade and non-safety-grade systems in the AP-600 will be subjected to high levels of thermal and mechanical loading. In view of the unique features of the plant design, it may be appropriate to require testing of components or systems that will be involved in severe accident mitigation or accident management. No specific vendor-sponsored testing related to severe accident performance has been identified at this time for AP-600. The staff will be evaluating the applicability of current industry-sponsored research to the AP-600 design, as well as determining where additional vendor-sponsored testing is appropriate. Specific issues include:

a. Debris Coolability

The proposed criterion for debris coolability for AP-600 is that at least $0.02 m^2/MWt$ of surface area should be allowed in the event of vessel failure and debris relocation to the reactor cavity. The industry is currently co-sponsoring a series of tests, (MACE) with NRC, DOE and several countries to investigate debris coolability for current generation plants. In the event of a core melt accident, the AP-600 design, with its lower core power density, would generate a larger volume of core debris per unit of thermal power, thus creating a deeper debris bed than expected for current plants. The applicability of the ACE/MACE data to the AP-600 design must be evaluated. It is recognized that there are differences between the experiments and the AP-600 design. The determination of the degree of applicability of these experiments therefore is critical in the assessment of this design and the determination of whether additional testing is needed. The method of determining applicability could use supporting analyses to aid in maximizing the usefulness of the existing program data to the AP-600. The specific approach will be the responsibility of the vendor.

b. Hydrogen Generation and Control

The AP-600 containment will include a hydrogen igniter system to deal with severe accident-generated hydrogen concentrations. Data related to optimized placement of the igniters and their performance are required for analytical assessment of severe accident containment performance.

c. Containment Performance in Severe Accidents

The AP-600 Passive Containment Cooling System (PCCS) must be capable of dealing with severe accident thermal loads, and the containment itself must withstand severe accident mechanical loads. Data are needed for model validation to permit analytical assessment of these containment performance issues. The method by which this assessment will be obtained could vary dependent on several factors. The data could vary dependent on several factors. The data could be generated by incorporation of changes into the PCCS test program, in Section II.E above, supporting the data base with appropriate analyses, or a combination of these two approaches. The specific approach will be the responsibility of the vendor.

d. Fuel Coolant Interaction (FCI)

The vendor's research program does not address the interaction between core debris and water either within or external to the vessel. The focus of this investigation should be both to explore the credibility of these events and to assure, for those credible events, that (1) a coolable configuration will be achieved; (2) fuel-coolant interaction (FCI) steam and hydrogen generation are accounted for; (3) containment response due to dynamic loads from an FCI are acceptable; and (4) FCI dynamic effects on accident progression are considered. Furthermore, the dynamic effect of corium spreading in the presence of a large water pool has not been completely substantiated by the existing experimental data.

The staff recommends that the testing and evaluations detailed above be performed. Further vendor-sponsored testing related to severe-accident issues may be identified as the staff continues its evaluation of the AP-600 design.