

WCAP-12965

WESTINGHOUSE PROPRIETARY CLASS 3

AP600
CERTIFICATION PROGRAM DESCRIPTION
WESTINGHOUSE ELECTRIC CORPORATION

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by

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1.0 INTRODUCTION

Purpose

The AP600 Certification Program Description addresses the major technical issues and dates when feedback is required on their resolution, nonpolicy issues, and programs where the AP600 program would benefit from early NRC Staff interaction. The schedule which will be followed for the AP600 review, design approval and certification is also identified. This description is intended to enhance the certainty and predictability of the certification assessment by providing for early consideration of all safety issues in the design development program. Timely feedback on AP600 issues is essential to avoid costly rework on the AP600 design and testing program.

Unresolved technical issues are being addressed through the ALWR Utility Requirements Document (URD) review. In the March 1, 1991 letter from Mr. W. R. Sugnet (EPRI) to Mr. Crutchfield (NRC), EPRI issued the list of technical issues central to the design of the passive plants. Westinghouse is supporting resolution of these issues, and their final disposition will be incorporated into the AP600 design thereby minimizing the unresolved technical issues prior to the AP600 Standard Safety Analysis Report (SSAR) submittal in June, 1992.

Background

The objective of the AP600 design and certification program includes completion of the detailed design, development, and safety evaluation of the AP600 plant, as required, to obtain NRC final design approval and design certification by the end of 1994, thus permitting commercial operation of the first plant by 2000. The AP600 licensing plan supports the overall program objectives by providing for early interaction between the NRC staff and the plant design activities to address safety and regulatory issues and establish up-front agreements on the safety assessment criteria and review processes. Key elements of the AP600 licensing plan include the following:

1. Establishing the certification program with the NRC staff early in the design process and prior to completion of detailed design.
2. Resolving key licensing issues in parallel with the ongoing AP600 design and development program and prior to completion of the final safety analysis.
3. Preparing a Standard Safety Analysis Report (SSAR) which is sufficiently detailed to permit staff safety evaluation as required for Final Design Approval and Design Certification.
4. Preparing a Probabilistic Risk Assessment (PRA) providing plant, containment, and site analyses for internal and external events and severe accident management assessment.

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5. Developing a program of Inspections, Tests, Analysis and Acceptance Criteria (ITAAC) which is sufficiently detailed to demonstrate that the remaining site specific design, along with subsequent procurement, construction, and startup testing, is completed in strict compliance with the licensing basis established for design certification.

The schedule for the DOE sponsored AP600 Design Certification Program is to have all certification activities completed by the end of 1994. The supporting activities for this schedule require submitting the SSAR, the PRA, and the ITAACs for NRC Review in 1992. The final design approval of the AP600 would be issued after 18 months of NRC review and the design certification after a rulemaking process that was estimated to take 12 months.

The format, content, and approach presented in the AP600 Certification Program Description is structured to achieve the following:

- o Maintain the Westinghouse AP600 design certification schedule of 12/94
- o Receive needed NRC feedback on the AP600 program
- o Meet the intent of the SECY-90-065 requirement that the ALWR URD review precede the review of any plant specific design information

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2.0 SCOPE OF APPLICATION

The AP600 application for design certification will include the information required by 10 CFR Part 52. The AP600 design documentation provided with the application for final design approval and design certification will be sufficiently complete to enable a safety evaluation of the AP600, including design basis accident analysis and severe accident review. The safety evaluations will be based upon design information and the standard site defined for the AP600 program.

The content of the AP600 application for design certification is addressed below.

Code of Federal Regulations

52.47(a)(1)(i)

The AP600 application for design certification will address the technically relevant information that is not site specific, as required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50, and its appendices, and Parts 73 and 100.

General Design Criteria

The General Design Criteria identified in 10 CFR 50, Appendix A will be addressed to identify how compliance will be achieved. If it is determined to be not applicable, exclusion will be addressed based upon it being not technically relevant or site specific.

Standard Review Plan and referenced standards

Each Standard Review Plan acceptance criteria will be evaluated to determine the AP600 design compliance. Those that are not technically relevant or site specific will be identified. Discussion will be provided for those to which the AP600 design takes exception.

Additionally, the SRP acceptance criteria will address compliance with the following associated sources of requirements:

- o Regulatory Guidance
- o Industrial Codes & Standards
- o Branch Technical Positions

TMI Requirements (10 CFR 50.34(f))

52.47(a)(1)(ii)

AP600 compliance with the technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) will be addressed.

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Site Parameters

52.47(a)(1)(iii)

The site parameters postulated for the design will be included in the design certification application. The design assumptions will be based on the identified parameters. Evaluations will be addressed in the applicable SSAR sections.

USIs and Medium and High Priority GSIs (NUREG-0933)

52.47(a)(1)(iv)

Based on the schedule for submittal of the AP600 design certification application, compliance with the technically relevant Unresolved Safety Issues and medium and high priority Generic Safety Issues that are identified in NUREG-0933 as of January 1, 1992 will be addressed.

Probabilistic Risk Assessment

52.47(a)(1)(v)

A probabilistic risk assessment will be included in the AP600 design certification application.

Inspection, Test, Analysis, and Acceptance Criteria

52.47(a)(1)(vi)

The AP600 design certification application will include the proposed inspections, tests, analyses, and acceptance criteria (ITAACs) which are necessary and sufficient to provide reasonable assurance that, if the ITAAC are performed and the acceptance criteria met, a plant which references the design can be built and will operate in accordance with the design certification.

Interface Requirements

52.47(a)(1)(vii)

The interfaces requirements to be met by the systems that are not safety related or are site specific will be identified. Sufficient detail will be provided to allow completion of the final safety analysis and design-specific PRA.

Interface Verification

52.47(a)(1)(viii)

The AP600 design certification application will include the inspection, testing, or analysis necessary to verify compliance with the interface requirements for the systems that are not safety-related or are site specific.

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Conceptual Design

52.47(a)(1)(ix)

The AP600 design certification application will include functional design information for those systems that are not safety related or are site specific. This is intended to aid the staff in its review of the final safety analysis and PRA, and to permit assessment of the adequacy of the interface requirements.

Specific Advanced Plant Standards

52.47(b)(2)(i)(A)

Safety feature performance demonstration and safety features interdependent effects acceptability will be addressed. The proof for each safety system design will be based upon the following:

- a. existing test, analysis, or experience
- b. specific supplementary analysis or test

Analytical methods verification will be addressed. Proof of the analytical methods verification will be based on the following:

- a. existing bases applicability, or
- b. specific supplementary method development and/or verification to be performed

Scope of Design

The scope of the AP600 design and development program provides an essentially complete plant design, which encompasses the entire power plant including all buildings, structures, systems, and components that significantly affect safety or impact the overall plant configuration. The level of design detail will be consistent with the safety significance of each system or component. Safety-related systems will be covered in detail. Functional design information and interface information will be included for systems that are not safety-related or are site specific.

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3.0 DESIGN CERTIFICATION PROGRAM SCHEDULE

The Westinghouse AP600 will be the first passive plant certified under 10CFR52. To meet the schedule of having the AP600 certified by the end of 1994, considerable effort and a continuing dialogue with the NRC is required. The AP600 design program includes developing a plant design with a level of detail consistent with the requirements of 10 CFR Part 52, completing the tests for demonstrating performance of safety features and the analytical models, completing the required documentation for the design certification submittal (SSAR, PRA and ITAAC), and completion of the certification process. Table 3.1 presents the projected AP600 design certification program schedule milestones. Westinghouse recommends this schedule be considered by the NRC in their development of the AP600 review. It is recognized that this schedule may change as a result of the information in SECY-91-161.

The dialogue with the NRC Staff was formally established when the AP600 Plant Description Report was submitted for NRC review in February 1989. The Commission has agreed to provide feedback on those aspects of the AP600 design that are not policy issues. Several meetings have been held with the Staff to review the AP600 test plan and how it supports the analytical models that will be used to perform the safety analyses that will be submitted in the SSAR. Additional meetings will be scheduled, as necessary, to ensure that the tests Westinghouse is performing mesh with those planned by the NRC to avoid duplication of effort.

Major policy issues related to the AP600 will be addressed via industry initiatives. This includes resolution of the technical issues central to design through the ALWR URD review. The dates when feedback is required on the technical issues are included in Section 10. Additionally, Westinghouse is participating in ongoing industry initiatives coordinated by NUMARC. NUMARC initiatives include addressing the National Environmental Protection Act (NEPA) concerns, defining the ITAAC process, and further definition of the rulemaking process.

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TABLE 3.1
AP600 DESIGN CERTIFICATION SCHEDULE

Standard Safety Analysis Report	6/92
- <u>W</u> Preliminary Safety Studies	12/90
- Level of Detail SECY-90-377	2/91
- <u>W</u> Final Safety Studies	3/92
- EPRI URD SER	3/92
Inspections, Test, Analysis, and	6/92
Acceptance Criteria Document	
- ITAAC SECY	5/91
- <u>W</u> ITAAC Guidelines	6/91
- <u>W</u> ITAAC Document	6/92
Probabilistic Risk Assessment Report	/92
- Preliminary PRA Studies	7/90
- NRC PRA Methods Review	6/91
- Complete PRA Calculations	3/92
- Submit PRA	6/92
NRC Safety Evaluation Report	6/93
- First ACRS Meeting	4/92
- First Round NRC Questions	9/92
- Responses	11/92
- Second Round NRC Questions	1/93
- Responses	3/93
- Draft SER	4/93
- ACRS Letter	6/93
Final Design Approval.	12/93
- Application for FDA	6/92
Design Certification	12/94
- Establish Hearing Procedures	12/91
- <u>W</u> Petition for Rulemaking	10/92
- <u>W</u> Draft DC Rule	10/92
- F.R. Notice of DC Rulemaking	12/92
- Appointment of Hearing Board	12/92
- Informal Hearing	12/93
- Hearing Board Preliminary Decision	2/94
- Formal Hearing (if required)	7/94
- Hearing Board Final Decision	10/94
- Final Notice of DC Rule	12/94

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4.0 LEVEL OF DETAIL PROVIDED IN DESIGN CERTIFICATION APPLICATION

A major consideration in deciding the appropriate level of design detail to be submitted or made available for audit, in support of a design certification application, is the sufficiency of the design information to enable the Commission to assure that the design can be constructed and operated in accordance with the Design Certification/Combined Operating License (DC/COL).

10CFR 52.47, "Contents of Applications", specifies the requirements for the contents of applications for design certification. In particular, 52.47(a)(1) requires the application to contain:

"(i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site specific."

In addition, 52.47(a)(2) addresses the level of detail which must be contained in the application or, if necessary for the NRC to make its safety determinations, be available for audit by the NRC Staff:

"The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination."

The information required to be submitted for a design certification under 10 CFR 52.47 will be presented in the format of a Standard Safety Analysis Report (SSAR) consistent with the format requirements of Regulatory Guide 1.70, Revision 3 and will address the functional and performance requirements of structures, systems, and components identified in the NRC Standard Review Plan (NUREG-0800). Consistent with NRC practice in past operating license reviews, the level of detail for Design Certification will vary from system to system depending on safety significance. Although a degree of general guidance can be given regarding the level of detail, the specific level of detail for each system can best be determined during the review process of the AP600, as expressly envisioned in Part 52. The final determination of what level of detail is appropriate for each system will be made by interaction between Westinghouse and NRC Staff during the SSAR review process.

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In this light, both the NRC and Westinghouse have substantial experience in the preparation and review of Safety Analysis Reports. There is a general understanding from regulatory requirements, Staff guidance, and broadly shared experience of the scope and level of detail sought in Safety Analysis Reports.

The Final Safety Analysis Report and the Staff's ensuing Safety Evaluation Report have constituted the primary bases for a safety determination and issuance of an operating license by the Commission. An important distinction imposed by Part 52, however, is the submittal of this information prior to plant construction.

It should be noted that the Part 52 process and applications thereunder, will result in a substantial increase in the level of detail being submitted to the NRC compared to what has historically been provided at the pre-construction stage under Part 50. Westinghouse will address USIs and GSIs, perform a PRA and document the resulting maintenance and reliability requirements, provide uncertainty analyses, and develop ITAACs for the AP600.

The depth of design detail submitted by Westinghouse for the AP600 design certification will be similar to that of a final safety analysis report at the operating license stage for a recently licensed plant (without the site-specific, as-built, and as-procured information that would not be available), plus ITAAC and PRA. Westinghouse will develop design criteria and bases, system descriptions, performance requirements, and component descriptions and characteristics in enough detail for the NRC to make its final conclusions on all safety questions. In terms of typical engineering design documentation, this will include system performance requirements, plant general arrangement and layout drawings, P&IDs, process flow diagrams and one-line electrical drawings. Additionally, general equipment locations, major pipe, duct and cable routing, QA program description, test and acceptance requirements, and pertinent design bases and analytical results and summaries will be included.

Information will be submitted consistent with the level of detail appropriate for the review contemplated by the SRP and relevant guidance and will be supplemented as needed by the question and answer process that has been successfully utilized by the Staff in the licensing of over 100 nuclear power plants. The Staff will have sufficient information to make the safety determinations required to support the approval of a design certification application. The difference in the varying levels of detail among systems would mainly be found in the specific descriptions of the physical attributes of individual system components. Whether or not such information is included, and to what degree, will depend on the safety significance of that given component to both the system and the plant as a whole based upon experience gained from previous operating reviews and our knowledge of NRC concerns on the AP600.

Performance requirements and design specifications for the structures, systems and components necessary to accomplish all safety functions will be described in the AP600 application. This will be sufficient to allow development of technical requirements for specific equipment procurement specifications and construction and installation specifications.

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Completed specifications including commercial information will not be available. Such "name plate" information (e.g., mounting bolt layouts, valve weights, handwheel locations and specific component data such as center of gravity criteria) is dependent upon actual procurement. To include such information in a design certification application would limit or preclude the ability of utilities to later obtain competitive bids for components and services. The emphasis should be on "performance requirements", as indicated in 10 CFR 52.47.

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5.0 AP600 COMPLIANCE WITH ALWR UTILITY REQUIREMENTS DOCUMENT

The development of the AP600 power plant design, which began in 1985, is being carried out as part of the United States ALWR program which is a cooperative effort sponsored by the U.S. Department of Energy and industry participants to establish the next generation ALWR power plant designs for the United States. Included in the ALWR program, under the leadership of the ALWR Utility Steering Committee, is the preparation of utility requirements for ALWR power plants. The requirements applicable to the AP600 power plant are included in Volume III of the ALWR Utility Requirements Document which was submitted for NRC review in September 1990.

The AP600 power plant is being designed to meet the requirements specified in Volume III of the ALWR Utility Requirements Document.

A process has been established within the ALWR program to address AP600 conformance to the URD. The AP600 design team will systematically review the baseline design for compliance with each utility requirement. Items of potential non-compliance and the underlying reasons will be identified and reviewed with the ALWR Utility Steering Committee to help the design team reach an ultimate compliance determination. As design changes are proposed in the course of AP600 design development, the changes will be reviewed for impact on utility requirements compliance and any change in compliance status will be reviewed with the ALWR Utility Steering Committee.

A summary of AP600 non-compliance issues, if any, will be developed and submitted to the NRC as a part of the review process.

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6.0 AP600 TEST PROGRAM

Introduction

Tests were conducted during the AP600 Conceptual Design Program (1986 through 1989) to provide input for plant design and to demonstrate the feasibility of unique design features. Tests planned for the AP600 design certification and detailed design program are designed to provide input for the final safety analyses, verify the detailed safety analysis models (computer codes), and provide data for final design and verification of plant components. The tests planned are listed in Table 6.1.

The AP600 tests related to the plant safety functions have been selected based on the plant features that are different from current PWRs and where directly applicable experimental data are not available. The tests are designed to provide simulation of plant features at a scale which is appropriate to demonstrate the phenomena being examined. To validate the computer models, these experiments will be modeled using the same computer codes used for plant analyses.

Testing of some plant component designs is required to verify their reliability and manufacturability. Other component tests will provide data for design optimization. The planned component design tests are discussed below.

6.1 AP600 Safety Related Tests

The AP600 design uses passive systems as the safety-grade means to mitigate the effects of Design Basis Events. The function of the passive safety systems is the same as that of the safety-grade active systems in a conventional PWR. Comparisons of the two plant safety system designs are shown in Table 6.2.

The AP600 safety related experiments are designed to meet several goals:

- 1) provide input to the final safety analysis
- 2) provide data on the passive safeguards systems to validate the safety analysis methods and computer codes
- 3) assess the design margin in the passive safety system performance

To accomplish these goals, a comprehensive experimental program has been developed which utilizes the available data from the NRC and industry light water reactor safety research programs as well as specific experiments which address the uniqueness of the AP600 system. The AP600 safety related test program utilizes component experiments to determine the transient behavior of the different AP600 safety system components such that computer models can be developed and verified. Additionally, tests are planned for the Passive Containment Cooling System (PCS) to develop and verify computer models.

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The range of plant conditions considered to establish the AP600 test program include: large-break loss-of-coolant-accident (LOCA), small-break LOCA, containment cooling, transient analysis, long-term core cooling and severe accident conditions. The uniqueness of the AP600 design was evaluated against current Westinghouse designs and safety-related data available in the literature (NUREG-1230). The results of this assessment were used to determine the data needs, defining the experiments required to support the AP600 safety analysis.

6.1.1 Large-Break LOCA

For large-break LOCA safety analysis the new features of the AP600 design include the addition of core makeup tanks (CMTs), gravity injection, and the use of canned-rotor reactor coolant pumps (RCPs). Two-phase pump flow data exists for Westinghouse designed pumps and others (NUREG-1230) that can be used to characterize the AP600 pumps. The core makeup tank, on the other hand, is unique to the AP600 design, so a specific test is planned for this component. Also, check valve operability with low differential pressure will be verified.

Core Makeup Tank Performance Test:

The purpose of this experiment is to verify the gravity drain behavior of the CMT over a full range of flowrates and pressures, and verify operation of the tank level indication which acts as a control for the Automatic Depressurization System (ADS). The water in the CMT drains by gravity head into the Reactor Coolant System (RCS) while steam is vented from both the cold leg and pressurizer to the top of CMT. This steam replaces the water drained from the CMT. Some of the steam condenses upon entry into the CMT and can impact the tank draining performance. The objective of the test is to verify that the tank will drain as calculated.

A one-sixth diameter and one-third height scale CMT will be constructed and instrumented to obtain the condensation rates within the tank to verify the computer model. The CMT water delivery and the performance of the level measurements will be examined over a full range of conditions.

Passive Safety Injection System Check Valve Tests:

One of the features of the AP600 design is the use of check valves to isolate passive systems from the RCS. Tests will be performed on these check valves to ensure their operability.

Tests will be conducted to demonstrate the capability of the Passive Core Cooling System (PXS) check valves to open under low pressure differential conditions which exist during gravity drain injection after having been held shut under high back pressure for an extended period of time and to measure check valve pressure drop at low flow velocities.

The check valve low differential pressure opening tests and flow tests will consist of duplicating normal pressure and temperature conditions on both the upstream and downstream sides of the check valves and times during which the valve has been held closed under high differential pressure. These temperature and pressure conditions will then be changed to model the plant response to various design basis events, simulating the expected flow conditions. Detailed data on valve opening and flow versus differential pressure will be obtained.

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6.1.2 Small-Break LOCA

For small-break LOCA safety analysis, the new features of the AP600 design are the CMT and the low ΔP check valves described above, and the ADS which depressurizes the primary system to near containment pressure.

The CMT provides injection flow to the reactor vessel at any RCS pressure. The planned core makeup experiments and check valve tests described above will duplicate the small-break conditions as well as the large-break conditions. The ADS provides controlled venting of the RCS to reduce pressure to assure transition to gravity driven injection from the in-containment refueling water storage tank (IRWST). Full scale tests will be conducted to demonstrate the performance of the ADS valves and spargers.

Automatic Depressurization System Hydraulic Test:

The purpose of this test is to simulate operation of the ADS to confirm the capacity and operability of the ADS valves and spargers, and to determine the dynamic effects on the IRWST structures.

A pressurized, heated water/steam source will be used to simulate the water/steam flow from the AP600 RCS during various stages of the ADS blowdown. The flow will be piped to a full sized sparger submerged in a quench tank simulating the IRWST. The piping will be designed to accommodate at least one of each ADS valve type. Instrumentation to measure water and steam flow rate, equipment dynamic loads, IRWST dynamic loads, and sparger/IRWST steam quenching will be provided.

6.1.3 Containment Cooling

A large number of experiments are planned to characterize the heat removal capabilities of the AP600 containment design. Several small scale experiments have already been performed during the AP600 conceptual design program to provide the database for models. These include the following:

- 1) Study of water films and wettability on steel plates simulating the containment exterior surface
- 2) Heated plate tests to examine the evaporating heat transfer of water from the steel surface of the containment and heat transfer with only air
- 3) Containment external cooling air flow path pressure drop tests to characterize the hydraulic losses
- 4) Steam condensation heat transfer experiments on a flat cool surface at different angles of inclination to simulate the condensation on the inside of the containment in the presence of noncondensable gases

In addition, tests were performed to examine the integrated behavior of the steam condensation on the inside, and the evaporative film cooling and air cooling on the outside of a pressure vessel. The cylindrical vessel used for this integral test was 3 feet wide and 24 feet high. These experiments will be continued during the detailed design and design certification program to examine the limits of coolability and the effect of cold weather conditions. Transient and steady-state tests will be run.

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Integral Containment Cooling Tests:

The purpose of these tests is to demonstrate the operation of the PCS over a range of operating conditions, including postulated severe accident conditions, and to evaluate the impact of low environmental temperatures on the operation of the PCS. These tests, in conjunction with completed conceptual design phase testing and the large scale containment test discussed below, will fully characterize the PCS design and performance. The PCS integral containment cooling test facility used to simulate severe accident pressure and temperature conditions has a working design pressure of 100 psig. Noncondensable gas concentrations will also be simulated.

Passive Containment Cooling System Heat Transfer Test:

A larger scale containment test will be performed to simulate the internal containment natural circulation and the heat transfer through the dome and side walls. This test will also help verify the containment analysis analytical methods.

This experiment examines the combined effect of natural convection and condensation on the interior of the containment while the exterior is cooled by film evaporation and air flow. This experiment will accurately model both the containment dome heat transfer and natural circulation inside the containment. It will complement the existing integral containment experiment which simulated the side wall condensation and evaporating film heat transfer.

A one-eighth diameter steel containment structure with exterior water film and natural circulation air cooling capability will be constructed. Detailed instrumentation will measure the condensation heat flux distribution, the resulting heat transfer coefficients, the air/steam mass ratios, and the resulting liquid film evaporation rates. Calculations of the current integral containment experiment and this containment test will be modeled using the detailed analysis model to verify the code and to show the scalability of the results.

Passive Containment Cooling System Water Distribution Test:

Since the distribution of the containment cooling water on the exterior shell of the containment is an important design item, a PCS water distribution experiment will be performed to examine and finalize the AP600 containment water distribution. The results will also provide input into the containment safety analysis computer codes for water coverage of the containment shell.

The test will be performed on a mock-up of a full scale sector of the containment dome. The AP600 water supply/distribution arrangement will be duplicated. Tests will then be conducted to demonstrate and measure the water spreading from the top center of the dome to the outer edges. Modifications to the water distribution arrangement will be made based on test results. All tests will be conducted with the surface coated with the prototypic AP600 containment coating. Detailed measurements of water film velocities and film thickness variation as a function of flow rate and radial distance on the dome will be obtained.

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Passive Containment Cooling System Wind Tunnel Test:

Since the containment relies on natural circulation of air to enhance evaporative cooling of the containment shell during a Design Basis Event, the local effects of the wind must be examined to ensure there are no adverse effects on the containment heat removal rates. A wind tunnel test will be performed to demonstrate that adjacent buildings, wind direction and plant site topography will not adversely affect natural circulation air cooling through the Shield Building and around the containment shell. The effects of high wind on the induced natural convection draft flow in the containment annulus and the resulting baffle static pressure distribution will also be examined.

An approximately 1/100 scale model of the entire AP600 plant will be constructed from plastic and instrumented with static and dynamic pressure taps. The model will be placed in a large scale wind tunnel and tested over a range of air speeds and directions to simulate different wind conditions. The results will be used to optimize the Shield Building air inlet and exhaust arrangement.

6.1.4 Non-LOCA Transient Analysis

The non-LOCA accidents will be evaluated using the same transient analysis methods used on existing Westinghouse PWR designs. However, since passive core cooling systems have been added to the AP600 design, models for those systems will be developed and added to the transient analysis codes. These models include a CMT model based on the CMT tests discussed previously, a passive residual heat removal (RHR) heat exchanger model, and a vessel mixing model.

Passive Residual Heat Removal Heat Exchanger Performance Test:

The AP600 has a passive decay heat removal system which replaces the active, safety-grade Auxiliary Feedwater System. The passive residual heat removal (PRHR) heat exchanger is located in the IRWST. This heat exchanger, which is connected to the primary side of the steam generator, removes decay heat energy to the IRWST water and depends only on natural circulation driving forces. If the IRWST water boils, the containment then becomes the heat sink through the PCS.

The PRHR heat exchanger test was designed to determine the heat transfer characteristics of the PRHR and the mixing characteristics in the IRWST. These key issues are instrumental in determining the heat exchanger size and configuration.

The test facility consists of three full-length heat exchanger tubes placed vertically in a cylindrical tank filled with water to simulate the AP600 IRWST. Under the AP600 conceptual design program, water at prototypic natural circulation and forced flow rates was run through the heat exchanger tubes at prototypic system pressure and temperatures. In the design certification program, tests will be conducted over an expanded range of tube flow rates and inlet temperatures. Additional data at intermediate temperatures will be taken to fully define the PRHR heat transfer correlation. Tests will also be conducted using a baffle to simulate the effect a second parallel row of tubes will have on heat exchanger thermal performance and tank mixing.

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6.1.5 Long-Term Core Cooling

Following a LOCA, the core cooling behavior of the AP600 is similar to that of an existing PWR. The main difference is the AP600 has more paths to vent steam from the core including the break, the pressurizer ADS valves, and the hot leg ADS valves. In existing PWRs, an active low-head safety injection pump provides flow to the cold legs of the reactor, first from the refueling water storage tank, and later from the reactor sump. The injected water cools the core by gravity driven flow from the downcomer to the core. A similar situation exists in the AP600 except that the water injected into the RCS comes from the IRWST which feeds into the reactor vessel downcomer. To demonstrate this passive method of long-term cooling, a long-term cooling experiment is planned.

Long-Term Cooling Test:

The purpose of this experiment is to demonstrate the operation of the long-term makeup path from the IRWST. In addition, this experiment will demonstrate subcooled water flow through the core to limit the long-term concentration of boric acid.

A scale model made of glass will be used to simulate the reactor vessel, steam generators, RCPs, IRWST, the ADS vent paths, the lower containment, and the connecting piping. The hot legs and cold legs will be simulated up to the CMT and pressurizer connections. Water will be the working fluid and the core will be simulated with electric heater rods scaled to match the core power levels consistent with the test scaling approach. The analysis methods used to determine the AP600 long term cooling will be compared to the experimental results.

6.2 AP600 Component Design Tests

The component tests will provide a larger database for design optimization during the detailed design of the plant. Several tests on individual plant components will also be performed to confirm their reliability or that their materials and fabrication methods meet ASME requirements.

Reactor Coolant Pump Flow Test:

Because of the direct effect of coolant flow on reactor performance and safety margins, it is essential that the reactor coolant pump hydraulic performance (head, flow, efficiency) be verified before AP600 design certification is granted. Established Westinghouse practice on new pump models is to perform a large-scale, low pressure, low temperature water flow test on a model of the complete pump hydraulics (impeller, diffuser, casing). This test will predict the head, flow, and hydraulic efficiency of the full scale pump. The pump flow test will be performed by the Westinghouse Electro-Mechanical Division. The pump hydraulic elements are scaled dimensionally at 58 percent of the full size reference pump.

Thermal Hydraulic Reactor Vessel Internals Test:

This test will be performed on an approximately one-seventh scale model of the reactor vessel and internals including vessel inlet and outlet coolant nozzles. The purpose of the test is to verify the hydraulic characteristics of

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the AP600 vessel and internals (e.g., flow distribution and pressure losses) including the AP600-specific design features which differ from previous standard designs.

Preliminary air model flow visualization tests will first be performed on a smaller scale model and will provide input into the design of the hydraulic test. Then, on the hydraulic model, testing will be performed to simulate the range of flow conditions expected in the actual plant, including a possible mismatch in the flow rates in the four inlet nozzles. During the test, it may be necessary to include several minor modifications to the internals to achieve the proper flow distributions. For example, it may be necessary to change the pattern or size of the holes in the bottom core support plate to optimize the flow distribution within the core from a thermal performance viewpoint.

Incore Instrumentation System Tests:

Systems similar to the AP600 top mounted fixed incore detector (FID) instrumentation have been demonstrated in operating plants, but a test is required to demonstrate that the system will not be susceptible to electro-magnetic (EM) interference from the nearby Control Rod Drive Mechanisms. It is also necessary to show that the design will not be subject to flow induced vibration which could lead to fretting wear and failure of the FID pressure boundary thimble.

The EM Interference test will be performed by mocking up instrument cables, bringing them into close proximity with an operating control rod drive mechanism, and measuring the resulting noise induced on simulated flux signals.

For vibration testing, a full scale FID pressure boundary thimble, external guard tube housing, internal guide tube and support column, spring loaded fuel nozzle insert guide, and the top fuel nozzle and fuel instrument thimble will be fabricated of actual specified materials. These will be housed in a low pressure/low temperature water pressure vessel and jacket which mocks-up the fuel and internals flow geometry. Pump flow at rated mass velocity and pump overspeed (approximately 115%) conditions will be provided to simulate the worst case flow induced vibration of the reactor for several hundred hours. Disassembly and inspection for wear will follow.

Steam Generator to Pump Material Welding Test:

The preferred pump casing material, because of its high allowable stress and excellent corrosion resistance, is a 13 chrome - 4 nickel alloy (SA487). Westinghouse has not previously welded this alloy to the low alloy carbon steel (SA508) material being used for the steam generator channel head or the 316 stainless steel discharge pipe material. A transition weld test may be needed to verify the welding procedures and weld properties.

Weld test samples of representative channel head nozzle, pipe and pump materials will be procured. These samples will duplicate the manufacturing process (forging and casting), heat treatment and thicknesses specified for the AP600 components. Trial welding procedures based on other transition welding experience will be developed. The as-delivered mechanical properties of the

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samples will be determined in a certified test laboratory, the samples will be welded using the range of procedures specified, then the welded test samples will be inspected and tested for soundness (ultrasonic test), dimensional change, and post weld mechanical properties.

Composite Concrete/Steel Structural Tests:

The purpose of these tests is to confirm the behavior of the concrete filled steel plate structures used in the modular construction of the containment interior structures and to develop and justify the criteria established for their design.

Simple beam and slab models will be constructed to represent typical design configurations used in the interior structural steel moduals. The prototype structures are typically 2 to 4 feet thick with one-half inch steel plates on each side. Since the test models will be approximately one-quarter scale, the exterior plates are one-eighth inch thick. The test models will be designed in parallel with design of the AP600 structures such that they investigate the critical design assumptions. Typical beams will be tested under axial and transverse loading. Typical slabs will be tested under in-plane and out-of-plane loading. Testing will include monotonic and cyclic loads representing seismic conditions.

I&C Class IE Cabinet Temperature Test:

This test will demonstrate the capability of the I&C components required for safety system actuation and reactor monitoring to operate for at least 72 hours with no reliance on normal HVAC for air cooling or circulation.

Cabinets containing typical I&C electronic components will be operated over the range of temperatures expected following a station blackout. Temperatures throughout the cabinet will be monitored to ensure that present component qualification temperature limits are not exceeded. Modifications to card orientations, card density, and cabinet air inlet and outlets will be made if required.

Reactor Coolant Pump/Steam Generator Airflow Test:

The airflow test will identify effects on the pump performance due to non-uniform channel head flow distribution, pressure losses of the channel head nozzle dams and pump suction nozzle, and will identify possible vortices in the channel head induced by the pump impeller rotation.

The air test facility will be an approximate one-half scale mockup of the outlet half of the channel head, the two pump suction nozzles, and two pump impellers and diffusers. The channel head tube sheet will be constructed from clear plastic to allow smoke flow stream patterns to be seen.

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High Inertia Rotor/Bearing Test:

A rotor, manufactured of depleted uranium clad with stainless steel, has been incorporated into the canned-motor RCP to provide the required flow coastdown performance for loss of flow transients.

Tests have been performed to verify manufacturability of the rotor, to accurately determine friction and drag losses, to verify the operating performance of the pivoted-pad bearing, and to develop a detailed quantitative knowledge of the factors influencing bearing design and performance. Tests will be performed to verify the drag losses of the rotor with the journal bearing located on the pump shaft. Approximately 1000 cycles of starts and stops will also be performed as a life test to demonstrate that the rotor will maintain its dimensional stability. These tests will be performed on the specially-constructed, full-scale rotor/bearing test rig.

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TABLE 6.1

AP600 DETAILED DESIGN AND DESIGN CERTIFICATION TESTS

SAFETY RELATED TESTS

LOCA MITIGATION

- o Core Makeup Tank Performance Test
- o Passive Safety Injection System Check Valve Tests
- o Automatic Depressurization System Hydraulic Test

CONTAINMENT COOLING

- o Integral Containment Cooling Tests
- o Passive Containment Cooling System Heat Transfer Test
- o Passive Containment Cooling System Water Distribution Test
- o Passive Containment Cooling System Wind Tunnel Test

NON-LOCA TRANSIENTS

- o Passive Residual Heat Removal Heat Exchanger Performance Test

LONG-TERM CORE COOLING

- o Long-Term Cooling Test

COMPONENT DESIGN TESTS

- o Reactor Coolant Pump Flow Test
- o Thermal Hydraulic Reactor Vessel Internals Test
- o Incore Instrumentation System Tests
- o Steam Generator to Pump Material Welding Test
- o Composite Concrete/Steel Structural Test
- o I&C Class 1E Cabinet Temperature Test
- o Reactor Coolant Pump/Steam Generator Airflow Test
- o Reactor Coolant Pump High Inertia Rotor/Bearing Test

TABLE 6.2

COMPARISON OF SAFETY-GRADE COMPONENTS AND SYSTEMS USED IN SSAR CHAPTER 15 ANALYSES

<u>ACCIDENT TYPE</u>	<u>CURRENT PWR-ACTIVE SAFETY GRADE COMPONENTS/SYSTEMS</u>	<u>AP600⁽¹⁾ PASSIVE SAFETY GRADE COMPONENTS/SYSTEMS</u>
Assured RCS Inventory For Small Leaks	- High Head Safety Injection Pumps	- Core Makeup Tanks
Small Break LOCA	- High Head Safety Injection Pumps - Cold Leg Accumulators	- Core Makeup Tanks - Automatic Depressurization - Accumulators - IRWST Gravity Drain
Large Break LOCA	- High Head Safety Injection Pumps - Cold Leg Accumulators - Low Head Safety Injection Pumps	- Core Makeup Tanks - Automatic Depressurization - Accumulators - IRWST Gravity Drain
22 Long-Term Cooling	- Low Head Safety Injection Pumps in Recirculation Modes - Containment Cooling (see below)	- IRWST Gravity Drain - Gravity Flow to Flooded Reactor Vessel Cavity - Passive Containment Cooling
Secondary Side Transients, Loss of Heat Sink	- Aux. Feedwater System, S/G PORVs - Low Head Safety Injection Pumps RHR Mode - Safety Grade Heat Sink (RHR HX, CCWS, ESWS, etc.)	- Passive Residual Heat Removal System - Passive Containment Cooling
Containment Cooling	- Safety Grade Sprays (Pumps) - Safety Grade Fan Coolers - Safety Grade Heat Sink (Spray HX, CCWS, ESWS, etc.)	- Passive Containment Cooling by Water and Air
Reactor Cool-down Transients	- High Head Safety Injection	- Core Makeup Tanks

(1) Note: The AP600 plant design includes active non-safety grade systems that can be used to mitigate the effects of all accidents.

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7.0 ANALYSIS METHODS

The AP600 design utilizes passive safety systems to provide decay heat removal, containment cooling, and core cooling in the unlikely occurrence of an accident coincident with a loss of all ac electrical power. The performance of the passive safety systems will be evaluated over a wide range of postulated accident scenarios. The passive safeguards systems provide the same functionality as the active systems on a conventional PWR as shown in Table 6.2. The only real difference is how the water or air is delivered to perform the protective function.

The AP600 passive safety systems rely on natural circulation cooling modes that are driven by buoyant forces resulting from fluid temperature differences. The analytical methods used to predict the performance of these systems will be verified against test data which simulate the same or similar buoyant/natural circulation situations. The AP600 experimental program, addressed in Section 6.0, as well as existing data from the Westinghouse NRC/industry light water reactor research program, will validate the analytical methods used to assess AP600 passive system performance.

In light of the increased desire for code accuracy, best estimate thermal-hydraulic methods, which give realistic predictions of system performance, will be used wherever possible. This analysis approach will also identify and quantify the margins available in the passive safety system design, giving added confidence in the design.

The following sections describe the analytical methods and related verification activities which will be used to calculate the AP600 design basis accidents safety performance for each major accident category.

The accident analysis methodologies that will be applied to the AP600 are summarized in Table 7.1.

7.1 Loss-of Coolant Accidents

7.1.1 Large Break

The AP600 design basis loss-of-coolant accident (LOCA) analysis will be performed using methods similar to those used for existing PWRs, the WCOBRA/TRAC code (Reference 1). WCOBRA/TRAC is a best estimate thermal-hydraulic systems code which has been used to analyze existing Westinghouse two, three, and four-loop PWRs with active systems. The WCOBRA/TRAC code will use the NRC approved best estimate Appendix K methodology (Reference 2). In applying the best estimate methodology, a significant effort will be made to quantify the computer code accuracy. Table 7.2 shows the different experimental tests which provide the basis for determining the code uncertainty. Also identified in Table 7.2 are specific tests which will be analyzed with WCOBRA/TRAC for the AP600 design.

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7.1.2 Small Break

The AP600 small-break LOCA analysis will be analyzed using the WCOBRA/TRAC code (Reference 1). WCOBRA/TRAC is a best estimate thermal-hydraulic systems code which has been used to analyze existing Westinghouse two, three, and four-loop PWRs with active systems for large break LOCA, and will also be used for the AP600 large break LOCA analysis. The small break LOCA analysis will be a best estimate analysis. Current analysis on the AP600 design using the more conservative Appendix K requirements in the NOTRUMP code (Reference 3) shows no core uncover for the small breaks which have been analyzed.

One new feature of the AP600 plant is the Automatic Depressurization System (ADS). If a small break would occur, draining the core makeup tanks (CMTs) would activate the ADS. The resulting rapid depressurization of the primary system would increase the rate of water addition into the vessel from the CMTs and accumulators.

Typically, PWR small break transients are calculated to occur primarily at high pressures (greater than 500 psia). With the addition of the ADS, the small break transient will now cover the full pressure range from limited steady state operation down to pressures as low as 25 psia. Therefore the existing small break analysis method must be validated for these low pressure applications. Table 7.3 shows the data validation basis for the WCOBRA/TRAC code under small break LOCA conditions. The additional validation against the FLECHT-SEASET natural circulation experiments at low pressure will verify the applicability of WCOBRA/TRAC at these low pressures.

The AP600 has passive safety systems such as the ADS and CMT. Validation of the WCOBRA/TRAC code on the experiments indicated in Table 7.2 as a part of the large break LOCA model development will ensure these passive features are adequately modeled.

7.2 Transient and Steam Generator Tube Rupture

AP600 plant transient analysis will be performed for all the typical PWR Class II, III, and IV transients which appear in a Chapter 15 SAR. The methods which will be used for the AP600 are the currently licensed transient analysis analytical tools; LOFTRAN, FACTRAN (Reference 12), TWINKLE (Reference 13), and where needed, NOTRUMP.

Since passive features have been included in the AP600 design, models for these systems will be incorporated into the analysis codes. These specific models are shown in Table 7.4, along with the source of validation from the AP600 experimental program as discussed earlier. For asymmetric transients such as steamline break, the mixing of the colder primary water from the faulted steam generator with the primary water from the intact loop will be simulated in the 1/7 scale reactor vessel hydraulic tests, see Section 6.2. Mixing factors will be determined from this experiment and will be used in the plant analysis. For feedline breaks or loss of heat sink, the passive residual heat removal (PRHR) heat exchanger immersed in the IRWST will remove decay heat, either with forced reactor flow, or natural circulation. The PRHR model will be added to the LOFTRAN systems model to analyze these transients. For events where emergency boration and Reactor Coolant System (RCS) inventory control is required, a CMT model will be added to LOFTRAN. Transient analysis models specific to the AP600 will be based on AP600 specific experiments.

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The Steam Generator Tube Rupture (SGTR) event will be analyzed using the LOFTTR2 (Reference 14) program. LOFTTR2 is a modified version of LOFTRAN with a more realistic break flow model, two region steam generator secondary side, and improved capability to simulate the operator actions during a SGTR accident. Modifications required to model the PRHR and the CMT will also be incorporated. If ADS actuation is predicted for a SGTR event, the transient responses after actuation of the ADS will be analyzed by the NOTRUMP code.

7.3 Containment

7.3.1 Containment Mass and Energy Release

The AP600 mass and energy release for containment integrity analysis will use both current methods and best estimate methods. Current methods using the SATAN (Reference 4), BASH (Reference 5), and TREAT (Reference 6) series of codes will be used for LOCA mass and energy release scoping studies. LOFTRAN (Reference 7) will be used to calculate main steamline break releases. The qualification of LOFTRAN is discussed in Section 7.2. In the second phase, the best estimate code WCOBRA/TRAC will be used to provide a more accurate calculation of the LOCA mass and energy release. The application of WCOBRA/TRAC to containment mass and energy release is also being developed for current plant designs.

7.3.2 Containment Integrity

The AP600 containment cooling design has undergone significant design scrutiny leading to a passive approach to this system unlike existing PWR plants. The containment will be passively cooled using a large (350,000 gallon) water supply and a natural draft air cooling flow path. The water will be distributed to the exterior surface of the containment steel shell for evaporation. The air flow up the outside of the containment also provides the air flow for evaporative cooling of the liquid films flowing downward over the containment surface. The design provides air cooling of the containment shell by natural circulation during both operating conditions and postulated accident conditions.

The flow on the inside of the containment is a buoyant-driven natural circulation flow with condensation occurring during postulated accident condition on the inside of the steel containment shell. The condensate is collected and channeled back to the IRWST.

To accurately predict the containment response for design basis accidents (LOCA and main steamline break), a two tiered analytical approach, supported by the AP600 experimental program discussed in Section 6, will be used. The COMPACT (Reference 8) containment analysis code and COBRA-NC (Reference 9) (GOTHIC) best estimate containment analysis code will be used to assess the detailed thermal-hydraulic performance inside and outside the containment. These codes have been assessed, and are continuing to be assessed against several different containment experiments as shown in Tables 7.5 and 7.6. As part of the AP600 program, there will be extensive analysis of the AP600 experiments validating these codes for buoyant, natural circulation thermal-hydraulic processes both on the inside and the outside of the containment. A reference AP600 plant calculation also will be performed with COMPACT and COBRA-NC (GOTHIC) to show the same thermal-hydraulic process observed in the AP600 specific containment experiments.

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The Westinghouse COMPACT code is currently being used for PWR containment integrity and sensitivity analyses. The initial AP600 reference design containment calculations (Reference 10) were performed with the COMPACT computer code which showed that the containment design criteria was not exceeded for a design basis accident and containment margin existed.

For short term containment subcompartment pressure loading analysis, the NRC approved TMD code (Reference 11), will be used. The passive design features of the AP600 do not impact the subcompartment analysis since peak pressures occur in the first several seconds of the transient.

7.4 Long-Term Cooling

In existing PWRs, long-term cooling is maintained by an active pump, heat exchanger, and water supply train which provide water to the RCS. The flow to the reactor coolant loop is forced. However, following postulated large LOCAs, the flow within the reactor vessel through the core is driven solely by the gravity head difference between the downcomer and the level in the core region.

The AP600 core cooling is similar to existing PWRs except the flow into the RCS is by gravity feed from either the IRWST or from the containment. Once inside the RCS, the flow within the reactor vessel through the core is driven solely by the gravity head difference between the downcomer and the level in the core region, just as in present PWR designs.

To model the long-term cooling portion of the transient, the initial calculations will first be performed using WCOBRA/TRAC, since this computational method will yield the most accurate and realistic assessment of the Passive Core Cooling System performance. As shown in Table 7.2, WCOBRA/TRAC will have been validated against low pressure natural circulation reflood experiments such as the Japanese slab core tests (SCTF) and cylindrical core tests (CCTF). In addition, there will be a scaled low pressure long-term cooling experiment performed using steam/water, with a heated core which can be used to assess the WCOBRA/TRAC code for this type of transient.

Once WCOBRA/TRAC has verified our understanding of this portion of the transient, a simplified analysis method will be developed and used to perform AP600 plant calculations for the SSAR and any additional sensitivity studies needed.

7.5 Radiological Consequences of Design Basis Accidents

The calculation of accident doses for the AP600 will be performed for a number of accidents including the following:

- Large Break LOCA
- Main Steam Line Break Outside Containment
- Steam Generator Tube Rupture
- Locked Reactor Coolant Pump Rotor
- Rod Ejection
- Fuel Handling Accident

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For the most part, the analyses will follow the methodologies that have been historically used for current operating plants. The radiological consequences of accidents will be calculated using atmospheric dispersion factors intended to encompass the majority of plant sites that might be considered. The AP600 design does not include active systems for the mitigation of activity releases but relies on passive design features to limit the consequences of the accidents. Part of the analytical approach that will be followed in the calculation of doses is that reductions will be made in some of the conservatisms that have been traditionally used.

One area of change is that the calculation of doses will utilize individual nuclide dose conversion factors from ICRP Publication 30 (Reference 14). The ICRP 30 values are more current values than those which have been used in past licensing analyses.

For accidents that involve a release of primary coolant, the primary coolant source term basis that will be used is that of 0.25 percent fuel defect level (instead of the current 1.0 percent). The normal operation upper limit for iodines will be reduced from the traditional value of 1.0 $\mu\text{Ci/g}$ Dose Equivalent (D.E.) I-131 to a value of which is consistent with the defined fuel defect level. The iodine spike limit will be similarly reduced from 60 $\mu\text{Ci/g}$ D.E. I-131 to a value which is consistent with the defined fuel defect level.

For accidents that involve a release of activity from the fuel clad gap, current methodology directs that it be assumed that ten percent of the core fission product inventory is in the gap. The gap fraction for presently operating plants is calculated to be substantially below this value. With the lower power density of the AP600 core (and associated lower fuel temperatures) there would be a lower rate of transport of activity into the gap. Consideration is being made of using a gap fraction lower than ten percent, consistent with results of the ANS 5.4 model.

7.5.1 Large Break LOCA

The calculation of doses will consider releases due to containment leakage and releases from an operating containment purge prior to its isolation. The calculation of doses resulting from the leakage of recirculating Emergency Core Cooling System fluid will not be performed since the AP600 design does not include any recirculation of post-accident fluids outside of containment.

The analysis will utilize the physically based source term defined by EPRI.

7.5.2 Steam Generator Tube Rupture

The analysis will take into consideration the potential for tube uncover. If tube uncover is forecast, the calculation of doses will use the methodology developed for the Westinghouse Owners Group.

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7.5.3 Fuel Handling Accident

The AP600 design includes no active system for the cleanup of releases from a fuel handling accident. The historical approach to the calculation of doses for the fuel handling accident is known to include very high levels of conservatism. We intend to reduce the level of conservatism in gap fraction and in the assumed peaking factors of the assembly described in the guidance provided in Regulatory Guide 1.25. The calculated doses will still include high levels of conservatism.

7.6 Probabilistic Risk Assessment

A probabilistic risk assessment (PRA) will be performed as part of the AP600 design certification as required by 10 CFR Part 52. The Level I PRA consists of an analysis of plant design, operation, and maintenance that focuses on accident sequences that could lead to severe core damage or containment challenge. The Level II PRA examines the plant accident mitigation features to determine vulnerabilities and containment performance in the event of a core damage accident. The Level III PRA analyzes the consequences to the general public in the event of a core damage accident with large radiological releases.

The major tasks involved in performing a PRA include the following:

- o Identification of Initiating Events
- o Quantification of Initiating Event Frequencies
- o Event Tree Modeling
- o System Analysis/Fault Tree Modeling
- o Data Analysis
- o Human Reliability Analysis
- o Dependent Failure/Common Cause Modeling
- o Fault Tree and Accident Sequence Quantification
- o Sensitivity and Importance Analysis
- o Containment Event Tree Analysis
- o Source Term Analysis
- o Consequence Analysis

The methodology employed in each of these major tasks is consistent with the methodology described in NUREG/CR-2300, "PRA Procedures Guide", and EPRI NP-2230, "EPRI Key Assumptions and Groundrules" (EPRI ALWR Requirements Document, Volume III, Chapter 1, Appendix A). These tasks are discussed in the following sections.

The methodology applied to the AP600 PRA is consistent with the methodology used by Westinghouse in performing PRAs for operating plants.

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7.6.1 Level 1 PRA

Identification of Initiating Events

Initiating events considered in the Level I PRA include both internal and external events. The internal events that occur as a direct result of equipment failure or as the result of errors while performing testing, maintenance, or other operator actions will be developed in the following manner:

1. Identifying the set of initiating events applicable to the AP600 plant from those listed in EPRI NP-2230,
2. Analyzing the failure of the auxiliary systems that might produce additional initiating events, and
3. Analyzing the failures of the AP600 innovative frontline systems that might produce new initiating events.

Many of the initiating events identified in this approach require the same response from plant systems and operators to avoid core damage, hence they can be grouped together. The initiators are grouped into the following five categories:

1. Manual Shutdown
2. Transients
3. Loss of Offsite Power (LOSP)
4. Loss of Coolant Accidents (LOCA)
5. Anticipated Transients Without Scram (ATWS).

Events whose consequences bound all the other events in the group will be considered as representative of each events group.

Initiating Event Frequency Quantification

The frequencies for all transients except loss of support systems and main steam line break events will be determined by a detailed review of INPC data on the operating experience of 51 PWRs spanning a total of nearly 220 reactor-years. Only events that occurred after commercial operation will be considered. The resulting frequencies will be adjusted to account for the greater availability factor expected for the AP600 plant. Adjustment factors will also be developed to account for the reduced number of loops present in the AP600 plant.

Event Tree Modeling

The event tree methodology will be used to represent the possible sequences of events following the initiating event groups defined in the analysis. Each event tree sequence depicts a combination of system and operator action successes or failures leading to either successful core cooling or to core damage.

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An event tree will be developed for each of the initiators and will be quantified using the Westinghouse fault tree linking code system, WLINK. This quantification process links each of the systems and operator actions required to respond to a particular initiator, appropriately modeling the interdependence of support systems and critical functions.

The modeling and quantification of the event trees results in the identification of core damage states, their associated probabilities, and their relative contribution to total core damage frequency. Plant damage states are defined as part of the Level I PRA and serve as a link between the core damage accident sequences and the containment response and source term analysis that are part of the Level II and III PRAs.

The internal accident sequences modeled describe multiple scenarios that could result in degraded core conditions. These scenarios are grouped to facilitate the containment analysis. For each initiating event, several event tree paths may be identical in terms of final plant conditions, since different combinations of success or failure of systems often yield the same general conditions.

In addition to categorization by plant conditions, event tree accident sequences are grouped into plant damage states based on:

- o Initiating event type (large LOCA, small LOCA, transient, containment bypass, etc.)
- o Timing of core damage (core damage occurs early or late)
- o Containment safety features operability

The plant damage states as defined in the Level I PRA include an assessment of containment systems (containment heat removal and radionuclide retention), containment bypass sequences, and containment impairment (loss of containment isolation).

System Analysis/Fault Tree Modeling

As part of the system analysis of the AP600 PRA, detailed fault trees will be developed for all the safety systems and several nonsafety systems whose operation could mitigate the effects of an accident in the AP600 plant.

The fault trees prepared for this analysis will be developed using the Westinghouse GRAFTER code system. In this system, the fault tree structure is input graphically and the basic event probabilities entered through the use of a master data bank. The GRAFTER code system is also used to quantify these fault trees.

Data Analysis

In general, the random failure data utilized in this analysis will be derived from the ALWR Requirements Document, Chapter 1, Appendix A. The logic and instrumentation failure data for the microprocessor-based components (circuit boards and cards) will be derived from Westinghouse data.

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Failure on demand probabilities from the ALWR generic data base will be used for components tested every month. For components with different test frequencies, the unavailability is considered proportional to the standby failure rate and the test interval, by neglecting the portion due to the stress on demand. The standby failure rates will be taken directly from the EPRI data base when available and will be derived from the demand unavailability values (of components tested monthly) in the generic data base when no standby failure rate was specified.

Human Reliability Analysis

Human reliability analysis (HRA) is used in the AP600 PRA to quantify the human error probabilities (HEPs) of operator actions and any other human errors that are modeled in the event trees or fault trees.

The HRA for the AP600 PRA is performed in accordance with the Human Reliability Analysis Guidebook and is consistent with the ALWR Requirements Document.

The human error analysis is performed according to the THERP, "Technique for Human Error Rate Prediction," methodology.

The master data bank provided in the AP600 Human Reliability Analysis Guidebook is used to assign probabilities to each operator action or subtask. The data bank contains mean values that are calculated from the median values in the THERP methodology by assuming a lognormal distribution.

Performance shaping factors (PSFs) are used to modify nominal HEPs to obtain more realistic modeling. Higher PSFs are applied in accident conditions where the operator stress level is great; lower PSFs are applied in testing or maintenance activities where it is thought to be highly unlikely that errors would occur. Dependencies among subtasks and between operators are considered in the THERP model. Cognition, omission, commission, and recovery errors are also modeled using THERP.

For the purpose of PRA evaluations, the generic Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs), developed for the low pressure reference plant design configuration, is applicable to the AP600. Therefore, procedural errors assessed on the AP600 are based on long or short procedures as determined by the number of steps in the WOG ERGs.

Dependent Failure Analysis

Dependent failures are those failures which defeat the redundancy employed in the design to improve the availability of some plant functions (such as coolant injection). A dependent failure arises from some causes that fail more than one system or more than one train of a system simultaneously or in a short interval.

AP600 CERTIFICATION PROGRAM DESCRIPTION

In the AP600 PRA, the following dependent failures will be analyzed:

- o Sequence functional dependencies
- o Inter-system dependencies
- o Dependencies due to human actions
- o Inter-component dependencies

Fault Tree and Accident Sequence Quantification

The methodology used in the quantification of the AP600 fault trees is the fault tree linking methodology. In this methodology, no support state event trees are developed and no support states are identified. Instead, those systems defined as support systems are analyzed using fault tree analysis. Fault tree linking uses the Westinghouse WLINK code system to link the fault trees created for the support systems and the frontline systems. The WLINK code accounts for the interactions and dependencies of the systems as it links the appropriate fault trees.

Sensitivity and Importance Analysis

Importance analysis is conducted to identify the important accident sequences, system failures, component failures, and human errors that contribute to the core damage frequency. Sensitivity studies are performed to investigate the variability in the core damage frequency and in the plant damage states as influenced by such factors as changes in the models, operator actions, initiating event frequencies, and possible plant design alternatives that decrease the unavailabilities of the important systems.

External Events Analysis

External event evaluation will include assessments of internal fires, flooding, seismic events, and other external events. The plant response to each external event will be assessed including evaluations leading to core damage frequency, releases, and consequences as necessary. The seismic assessment will be based on generic fragilities with subsequent detailed assessment of the critical components. The site location for the external event assessment will be chosen consistent with the ALWR generic site criteria.

7.6.2 Level II PRA

The Level II PRA will consist of the following three tasks:

1. Containment Performance Characterization
2. Containment Event Tree Analysis
3. Source Term Analysis

Containment Performance Characterization

The containment performance characterization includes identifying and evaluating the likely containment challenge modes for severe accident loadings, performing a structural capability assessment, and estimating a mean containment failure pressure.

AP600 CERTIFICATION PROGRAM DESCRIPTION

The potential containment challenge modes applicable to the AP600 will be identified through a review of the design versus the potential failure modes identified in NUREG/CR-2300 as well as direct containment heating and thermal attack of containment penetrations.

The structural capability assessment will be performed to determine pressure capability for the containment shell, piping penetrations, equipment hatches, electrical penetrations, etc. Existing data on structural capability from other studies and the NRC-sponsored Containment Performance Program will be used.

Containment Event Tree Analysis

This task includes developing an AP600 specific Containment Event Tree (CET), estimating the magnitude of containment challenge, timing of containment challenge for the representative sequence analyzed for each plant damage state, and quantifying the CET by assigning branch point probabilities.

The CETs will be developed in a manner consistent with the guidance in NUREG-1335. Some severe accident phenomena may also be represented in the CET top event nodes.

For each of the representative sequences analyzed using the CET, the applicable potential containment challenge mechanisms will be identified and the magnitude of the containment challenge estimated for the purpose of determining its timing and location. The MAAP code will be used for predicting the containment thermodynamic behavior during a sequence, including the effects of the applicable phenomena that are potential containment challenge mechanisms.

Source Term Analysis

Source term analysis includes defining AP600 specific radionuclide release categories, identifying the release category for each CET end state, estimating the release fractions of the risk-significant core fission products and performing sensitivity analyses of the results.

Source term characteristics will be determined for a single representative sequence from each release category. The MAAP code will be used to determine the source term for each representative sequence. Sensitivity analyses will be performed to estimate the effects of key analysis assumptions on timing of containment challenge and the magnitude of the source term.

MAAP Computer Program

The version of the MAAP code to be used for the study will depend on the version available to the AP600 program consistent with schedule for completion of the Level II PRA. The version currently projected for use is MAAP 4.0. However, if the MAAP 4.0 release is delayed by unforeseen problems, then MAAP 3.0B will be used. The MAAP 3.0B analysis may be supplemented by MAAP 4.0 analyses, where needed, after MAAP 4.0 is available.

AP600 CERTIFICATION PROGRAM DESCRIPTION

The models for MAAP that reflect the AP600 design features such as Passive Containment Cooling and In-Containment Refueling Water Storage Tank will be validated based on the AP600 testing program results in a similar manner to the validation for other thermo-hydraulic codes being used for licensing analysis.

7.6.3 Level III PRA

The objective of this effort is to provide an estimate of the offsite consequences associated with severe accident fission product releases from the AP600. The fission product characteristics defined in the Level II portion of the PRA will be used. The consequence estimates will be developed using a representative site to be chosen based on the likelihood of location of a plant at such a site.

The consequence analysis will be performed using the MACCS computer code. The MACCS code estimates the peak and mean consequences and the cumulative distribution functions (CCDF) by performing atmospheric dispersion simulation of the fission products released. The consequence indices to be investigated are the acute fatalities, latent fatalities, and the population dose (man-rem). The investigation will include the offsite dose levels as a function of distance from the plant for each of the principal exposure pathways (e.g., direct exposure from cloud passage, inhalation during cloud passage, and ground deposition). The consequence analysis will be performed assuming that no offsite emergency planning activities are implemented.

AP600 CERTIFICATION PROGRAM DESCRIPTION

7.7 REFERENCES

- 1) Hochreiter, L. E., Schwarz, W. R., Takeuchi, K., Tsai, C. K., and Young, M. Y., "Westinghouse Large Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation," WCAP-10924-P-A, Volume 1, Revision 1, December 1988 (Proprietary).
- 2) "Emergency Core Cooling Systems; Revisions to Acceptance Criteria," Federal Register, Volume 53, No. 180, pp 35996-36005, dated September 16, 1988.
- 3) Lee, N., Rupprecht, S. D., Schwarz, W. R., and W. D. Tauche, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985 (Proprietary).
- 4) Bordelon, F. M., Kelly, R. D., Massie, Jr., H. W., Muench, R. A., Spencer, A. C., and Young, M. Y., "Satin VI Program: Comprehensive Space-Time Dependent Analysis of Loss of Coolant," WCAP-8302, June 1974.
- 5) Besspiata, J. J., Kabadi, J. N., Yeh, H. C., and Young, M. Y., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, WCAP-10266-P-A, Rev. 2, March 1987 (Proprietary).
- 6) Frantz, E. R., Huang, P. H., and Ofstun, R. P., "Comparison of the TREAT and NOTRUMP Small Break LOCA Transient Results," WCAP-11232, September 1986 (Proprietary).
- 7) Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.
- 8) Tsai, S. S., et al., "COMPACT: A High-Speed Computer Code for Multi-Compartment Containment Transients Analysis," WCAP-10361, Rev. 1, June 1987.
- 9) Wheeler, C. L., Thurgood, M. J., Guidotti, T. E., and DeBellis, D. E., "COBRA-NC: An Analysis of Nuclear Reactor Components," NUREG/CR-3262, May 1986.
- 10) DE-AC03-90SF18495, "AP600 Plant Description Report" prepared for U.S. Department of Energy San Francisco Operations Office by Westinghouse Electric Corporation, December 1990.
- 11) "Ice Condenser Containment Pressure Transient Analysis Methods," WCAP-8077, March 1973.
- 12) Hargrove, H. G., "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO₂ Fuel Rod," WCAP-7908-A (Non-Proprietary), December 1989.
- 13) Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary), and WCAP-8028-A (Non-Proprietary), January 1975.
- 14) International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30.

AP600 CERTIFICATION PROGRAM DESCRIPTION

TABLE 7.1

SUMMARY OF AP600 ACCIDENT ANALYSIS METHODOLOGIES

LARGE BREAK LOCA

- o W COBRA/TRAC Thermohydraulics/Rod Heatup

SMALL BREAK LOCA

- o WCOBRA/TRAC Thermohydraulics/Rod Heatup

TRANSIENTS

- o LOFTRAN
- o FACTRAN
- o TWINKLE
- o LOFTTR2 SGTR response
- o NOTRUMP SGTR with ADS

CONTAINMENT

- o WCOBRA/TRAC M&E LOCA
- o LOFTRAN M&E SLB
- o SATAN M&E Short Term
- o COMPACT Containment response
- o GOTHIC(COBRA-NC) Containment response
- o TMD Containment Response - Short Term

LONG TERM COOLING

- o WCOBRA/TRAC Thermohydraulics

OFFSITE DOSES

- o NOTRUMP M&E Rod Ejection

AP600 CERTIFICATION PROGRAM DESCRIPTION

TABLE 7.2

WCOBRA/TRAC VALIDATION FOR LARGE BREAK LOCA

HEAT TRANSFER TESTS

- o W/NRC FLECHT-SEASET Reflood
- o W/NRC FLECHT Cosine Reflood
- o W/NRC FLECHT Skew Reflood
- o W G-2 Reflood
- o KFK FEBA Reflood
- o W G-1 Blowdown
- o W G-2 Blowdown
- o ORNL Film Boiling

CLADDING MATERIALS/NUCLEAR ROD TESTS

- o NRU Reflood
- o NRU Materials

FLUID MECHANICS TWO PHASE FLOW TESTS

- o CREARE 1/15, 1/5 Scale ECC Bypass
- o UPTF ECC Bypass Cold Leg Injection
- o 1/14, 1/3 Scale Cold Leg S/W Mixing
- o UPTF Cold Leg S/W Mixing
- o APWR Two Phase Pressure Drop
- o MARVIKEN Critical Flow Tests
- o UPTF Upper Plenum De-entrainment
- o * UPTF Direct Vessel Injection ECC Bypass Experiment
- o * CMT Test Analysis
- o * ADS System Behavior/Response/Critical Flow

WCOBRA/TRAC SYSTEMS EFFECTS TESTS VALIDATION

SYSTEM RESPONSE TESTS

- o LOFT L2-2, L2-3, L2-5, LB-1
- o Semiscale MOD 3 Series 7
- o CCTF Cold Leg Injection Experiments
- o ** CCTF Direct Vessel Injection Experiments
- o SCTF Cold Leg Injection Experiments
- o ** Oregon State University AP600 Long Term Cooling Test

*AP600 Specific Validation

**AP600 Long-Term Cooling Experiments

AP600 CERTIFICATION PROGRAM DESCRIPTION

TABLE 7.3

WCOBRA/TRAC SMALL BREAK LOCA VERIFICATION

CORE HEAT TRANSFER TESTS

- o W G-1, G-2 Level Swell

FLUID MECHANICS AND TWO-PHASE FLOW TESTS

- o 1/3-Scale Loop Seal Flooding Experiments
- o 1/3-Scale Hot Leg Flow Regime Experiments
- o * CMT Test Results
- o * ADS Critical Flow and System Results

SYSTEMS RESPONSE TESTS

- o LOFT Small Break Test Series
- o * FLECHT-SEASET Low Pressure System Effects

*AP600 Specific Validation

AP600 CERTIFICATION PROGRAM DESCRIPTION

TABLE 7.4

VALIDATION FOR TRANSIENT ANALYSIS MODELS

REACTOR COOLANT PUMP

- o RCP High Inertia Rotor/Bearing Tests
- o RCP Flow Model Tests

PASSIVE RHR MODEL

- o PRHR System Tests

CORE MAKEUP TANK MODEL

- o CMT Flow Behavior and Level Instrumentation Tests
- o PSIS Check Valve Tests

VESSEL HYDRAULIC MODEL

- o 1/7-Scale Vessel Internals Tests

AP600 CERTIFICATION PROGRAM DESCRIPTION

TABLE 7.5

COMPACT CONTAINMENT INTEGRITY AND HYDROGEN TRANSPORT VALIDATION TESTS

SHELL EVAPORATIVE HEAT TRANSFER AND CONDENSATION TESTS

- o * W AP600 Heated Plate
- o * University of Wisconsin Condensation Experiments

INTEGRAL SYSTEM TESTS

- o Carolinas Virginia Tube Reactor Containment Test
- o Battelle-Frankfurt (CASP-1)
- o Lucas heights (CASP-3)
- o Karlsruhe HDR (ISP-16, ISP-23)
- o * PCCS Integral Test Facility
- o * AP600 PCCS Large Scale Facility

HYDROGEN DISTRIBUTION TESTS

- o Karlsruhe HDR (Post ISP-23)
- o * AP600 PCCS Large Scale Facility

* AP600 Specific Validation

AP600 CERTIFICATION PROGRAM DESCRIPTION

TABLE 7.6

GOTHIC CONTAINMENT INTEGRITY AND HYDROGEN TRANSPORT VALIDATION TESTS

SHELL EVAPORATIVE HEAT TRANSFER AND CONDENSATION TESTS

- o * W AP600 Heated Plate
- o * University of Wisconsin Condensation Experiments

INTEGRAL SYSTEM TESTS

- o CSNI Standard Problems (No. 2 and 3)
- o Battelle Frankfurt C and D Series
- o Karlsruhe HDR Steam/Water Blowdown (V43 and V44)
- o Karlsruhe HDR Fire Experiments
- o Karlsruhe HDR Natural Circulation
- o Marviken Containment Response
- o Carolinas Virginia Tube Reactor
- o LACE
- o * PCCS Integral Test Facility
- o * AP600 PCCS Large Scale Facility

HYDROGEN DISTRIBUTION TESTS

- o Karlsruhe HDR (6, 12, and 15)
- o Battelle Frankfurt
- o * AP600 PCCS Large Scale Facility

* AP600 Specific Validation

AP600 CERTIFICATION PROGRAM DESCRIPTION

8.0 INSPECTIONS, TESTS, ANALYSES AND ACCEPTANCE CRITERIA

The AP600 application for a Design Certification rulemaking will include proposed means of assuring that construction conforms to the design and to allow the NRC Staff to reach a final conclusion on all safety questions associated with the design. The requirements of 10 CFR Part 52 are:

"Proposed tests, inspections, analyses, and acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections and analyses are performed and the acceptance criteria met, a plant which references the design is built and will operate in accordance with the design certification." (10 CFR 52.47(a)(1)(vi))

The method to be used for verification of interface requirements for those portions of the plant for which the application does not seek certification will also be included as part of the proposed tests, inspections, analyses, and acceptance criteria details submitted with the application for Design Certification.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) documents will be prepared for the AP600 to substantiate the adequacy of the systems, structures, components, and commodities to accomplish their intended safety functions.

Westinghouse will utilize the ITAAC approach developed by the NUMARC Task Force on Inspections, Tests, Analyses and Acceptance Criteria. This approach is being discussed with the NRC. This approach adopts a concept of a two tier design certification rule. The ITAAC provisions will be contained in the first tier of the Design Certification rule. These first tier elements can not be modified by any subsequent COL applicant/holder without a Commission authorized exemption or amendment to the rule.

The AP600 ITAAC development will begin with a definition of those systems, structures, components and commodities necessary to substantiate or support safety functions. The results of previous industry programs such as the Vogtle Unit 1 Readiness Review (NUREG-1278, September 1987) will be reviewed to help identify necessary systems, structures, components, and commodities subject to ITAAC provisions. Safety design bases for the systems, structures, components and commodities will be specified. The safety design bases are those criteria that are necessary and sufficient for the structures, systems, components, or commodities to perform their intended safety functions. These criteria are derived from existing requirements and regulations such as the General Design Criteria, Safety Criteria (e.g., 10 CFR 20, 10 CFR 50, Appendix E, etc.), Standard Review Plan (NUREG-0800), Industry Codes and Standards/Regulatory Guides, and other regulatory sources.

The safety design bases criteria will be evaluated to define the best methodology to be applied to demonstrate that the acceptance criteria have been satisfied. It is expected that most of the information necessary to demonstrate conformance with specified safety design bases will be provided in

AP600 CERTIFICATION PROGRAM DESCRIPTION

the text, drawings, tables, and associated information contained within the submitted AP600 Standard Safety Analyses Report. However, some aspects of the structures, systems, components and commodities will need to be field demonstrated to verify confirmation of design assumptions.

The ITAAC documents will consist of five sections. The first section will be an identification of the system, structure, component or commodity safety functions. The second section will contain a brief description of the system, structure, component or commodity. The third section will be a definition of the Acceptance Criteria necessary to be demonstrated. The fourth section will be a description of those inspections, tests, and analyses considered necessary to demonstrate conformance with the Acceptance Criteria. The fifth section identifies Validation Attributes that are readily verifiable to demonstrate conformance with a specified Acceptance Criteria. If Acceptance Criteria are sufficiently demonstrable as defined, corresponding Validation Attributes are not necessary and this section is not included.

The Acceptance Criteria will consist of those performance, functional, or physical characteristics that must be demonstrated as having been met by the system, structure, component, or commodity at the appropriate stages of construction, installation, or pre-operational testing. They will be a descriptive specification of demonstrable characteristics necessary and sufficient to substantiate the ability of the structure, system, component or commodity to perform intended safety functions. The Acceptance Criteria specified must be directly related to the safety design bases.

Should an Acceptance Criteria not be readily verifiable, a corresponding Validation Attribute will be provided which is readily measurable, or otherwise verifiable. Validation Attributes provide a technically sound and practical link between the quality assurance program and the Acceptance Criteria in those situations where, because of the nature of certain Acceptance Criteria, direct verification by field test or inspection is not practical. Validation Attributes are the way in which functional construction, component installation, performance parameters, materials and/or analytical characteristics of the plant can be confirmed. Validation Attributes would confirm, for example, analytical model results approved at the design certification stage by verifying critical parameters in the as-built structure.

The NRC presently has developed a comprehensive inspection program designed to provide reasonable assurance at the end of the construction process that a facility has been constructed, and will operate, in conformance with the Safety Analyses Report. It is expected that this inspection program will be utilized with some additional checks, and with the NRC providing for documentation relative to their extent of involvement with the construction verification processes. The NRC will witness relevant events that demonstrate field verification of ITAAC provisions.

ITAACs to be developed will be defined during the design process. All safety related systems will have corresponding ITAACs. Additional ITAACs for systems, structures, components and commodities which are not safety related will be developed as appropriate to demonstrate that the interfaces with safety systems have been properly addressed in the design.

AP600 CERTIFICATION PROGRAM DESCRIPTION

9.0 SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

A recent court decision has indicated that the NRC must also consider design alternatives in connection with NRC considerations of environmental matters under the National Environmental Protection Act, NEPA, even if those design alternatives do not need to be considered under the Atomic Energy Act. The intent of Part 52 was to preclude design considerations after a design has been certified.

While Part 52 has no specific requirement to address environmental impacts, Westinghouse believes treatment of Severe Accident Mitigation Design Alternatives (SAMDA) is an important issue that is best addressed early in the design and licensing processes of a nuclear power plant. By considering and resolving SAMDAs early in the design process, the potential for future licensing problems and costly design changes after design certification should be effectively eliminated.

The AP600 Standard Safety Analysis Report and Probabilistic Risk Analysis will be submitted to the NRC for review in June, 1992. We are now in the process of preparing these two documents. It is our intent to address the SAMDA issues as early as practical. Without knowing which of the options being considered by the industry will achieve finality on the SAMDA issues, or what other options might be available to achieve finality, it is premature for us to commit to a specific method of addressing them.

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.0 AP600 CERTIFICATION ISSUES

The following section presents those unresolved technical issues that are relevant to the AP600 design. Technical issues which are not fully resolved will be addressed through the ALWR Utility Requirements Document (URD) review. In the March 1, 1991 letter from Mr. W. R. Sugnet (EPRI) to Mr. Crutchfield (NRC), EPRI presented the list of technical issues central to the design of the passive ALWRs. The issues discussed in this section are consistent with those presented by EPRI in their March 1 letter. Westinghouse will support resolution of these issues, and the final disposition will be incorporated into the AP600 design thereby minimizing the unresolved technical issues prior to the AP600 Standard Safety Analysis Report (SSAR) submittal in June, 1992.

The AP600 is being designed in accordance with the positions stated in the writeups. If the industry position is changed at some later date, there will be either a disruptive effect on the AP600 design effort due to a need to modify the design or the end cost of the plant will have been driven up by carrying design features that were not necessary to ensure the safety of the plant and protection of the public. Dates by which NRC feedback on an acceptable resolution of these approaches is required are supplied. If a change to the plant design is required past the identified need date, we anticipate substantial affect on the AP600 design cost and schedule. Additionally, resolution that is not consistent with the current industry position could result in a substantial increase in the cost of an AP600 or an increase in the construction schedule time.

References to the ALWR URD are included as appropriate as a cross reference to the EPRI submittal.

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.1 Regulatory Treatment of Passive Safety Systems

10.1.1 Passive Safety Systems

Reference: ALWR URD, Vol. III, Chapter 5, Section 1.2.1

AP600 Position:

A primary objective of the AP600 design is to simplify the plant by using passive systems to replace all the active engineered safety systems presently used in pressurized light water reactors. Systems necessary to assure the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; and the capability to prevent or mitigate the consequences of design basis accidents which could result in off-site exposures in excess of regulatory limits will be classified as safety-related. These systems utilize either natural forces such as gravity or stored-energy.

The AP600 design is primarily passive in nature. There are no safety related pumps. Actuation of the engineered safeguards features requires only that a small number of DC powered valves change state. The Passive Containment Cooling System (PCS), Core Makeup Tanks (CMT), Accumulators, Passive Containment Spray System, In-Containment Refueling Water Storage Tank (IRWST) and emergency main control room HVAC are powered either by gravity or stored gas. There is no class 1E AC electrical power required for the AP600, hence no safety grade emergency diesel generators are included in the design. Emergency power is supplied from batteries with a 72 hour capacity.

The AP600 does have non-safety ac-powered pumps and equipment that are used for normal operations. These can also be powered from two non-safety diesel generators provided for owner's investment protection in the event of a loss of offsite power. In the event of an accident, these systems can be used to supplement the passive safety systems although the AP600 is designed to meet the intent of the NRC requirements in response to design basis events without reliance on non-safety systems.

This passive design philosophy is fundamentally different from past practices and, in some cases, requires staff evaluation of compliance with the intent of regulations and regulatory guidance. The NRC staff has expressed the need to adequately assess the reliability of passive safety systems, the extent to which the passive systems and the non-safety systems contribute to the overall reliability of protection against accidents, and the benefits of the simplification that accrue due to the use of passive safety systems.

Westinghouse believes that the combination in the AP600 design provides a high degree of defense in depth. We expect to show that the core melt frequency for the AP600 is quantitatively equivalent to that for a well designed evolutionary ALWR plant with active 1E safety systems. This level of risk will be attained with systems that are significantly simpler to construct, maintain and operate. This simplicity reduces dependence on operator action while providing increased diversity from the combination of active and passive systems.

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A significant AP600 safety advantage is derived from the ultra high containment cooling reliability provided by the simplicity of the passive containment cooling system. This advantage is reflected in a much lower risk of a significant offsite radiation release for AP600 as compared to an ALWR with 1E active systems.

Changes to the AP600 to make the active systems safety-grade would significantly increase the cost and complexity of the plant. It would also make operation and maintenance more complex. The benefits of the combination passive/active systems are achieved in the design as it now stands.

Need Date to Support Design: Now

10.1.2 Regulatory Treatment of Non-Safety Systems

Reference: None

AP600 Position:

In addition to the safety systems which meet licensing requirements for design basis events, the AP600 has active systems which could function to prevent or mitigate accidents. These active systems are installed for normal operation and are designed for high reliability.

The NRC has identified the regulatory status of these non-safety systems as an "emerging technical concern" in SECY-90-406.

Westinghouse believes the combination of passive safety systems and active normal operating systems is a key strength of the AP600. The passive safety systems are designed to meet all regulations during design basis accidents with no credit from the highly reliable active systems. However, the use of the non-safety grade active systems, with appropriate reliability factors, has been shown to improve the core melt frequency in the preliminary PRA.

Performance guidelines will be established for plant operation to ensure that the assumptions of reliability made in the PRA are valid. Any requirement for non-safety systems or their supporting systems to become safety grade or subject to rigid operational controls (i.e., Technical Specifications) is not necessary for safe, reliable operation.

Traditionally, non-safety systems have been credited in certain safety and regulatory processes: Probabilistic Risk Assessment; Severe Accident Mitigation; Station Blackout; Emergency Response and others. Non-safety systems have generally not been given credit for licensing design basis accident analyses and no such credit is being requested for the AP600.

Credit for non-safety systems is being sought where it has been used before: PRA, Severe Accident Mitigation, etc. This limited credit would allow systems classified as "non-safety" to retain that classification.

Need Date to Support AP600 Design: Now

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.2 Severe Accident

10.2.1 Containment Performance

Reference: ALWR URD, Vol. III, Chapter 5, Section 6.6.2, "Containment Performance"

AP600 Position:

The Commission has formulated safety goals against which the overall risk to the public resulting from plant operation can be compared.

The goal for the AP600, which is consistent with the ALWR URD, is more conservative than the Commission's policy and the Staff's proposed implementation guideline. The ALWR URD requires that the offsite dose consequences be less than 10^{-6} /yr frequency exceeding 25 Rem at the site boundary.

Meeting the industry goal provides a substantial margin in the level of public protection. In addition to this probabilistic goal, industry has established a set of deterministic requirements for a reliable and rugged containment to ensure defense in depth regardless of the quantified probability of adequate core cooling. These additional, specific requirements governing the minimum containment mitigation capability have been established in the ALWR URD based on realistic assessment of the threats to containment integrity posed by representative core damage accidents.

For similar reasons, the NRC is evaluating an additional implementation guideline associated with containment performance. The containment performance guideline being considered is a conditional containment failure probability (CCFP) of less than .1 or some deterministically established containment performance requirements which achieve a similar objective as the CCFP.

Design features of the AP600 associated with severe accident prevention and mitigation will effectively provide the alternative, deterministically-based containment performance criteria. Notwithstanding an extremely low probability of core damage for the AP600, these requirements mandate a reliable and rugged containment with design features explicitly directed both at limiting the magnitude and likelihood of specific severe accident challenges and at assuring the capability of the containment to accommodate the risk-relevant challenges that remain.

In addition, as noted in section 10.4.1, Westinghouse is working with EPRI in a program to reduce the magnitude of fission product releases resulting from representative AP600 core damage events such that the protection action guides are not exceeded at the site boundary. This goal is being established to provide a basis for simplification of emergency planning, eliminating the need for early notification and evacuation planning drills.

To attain these goals, the AP600 containment will be designed to assure that if a core damage event were to occur, the containment would be sufficiently rugged to withstand the expected challenges associated with such an event.

AP600 CERTIFICATION PROGRAM DESCRIPTION

Need date to Support AP600 Design: Now

10.2.2 Core Debris Coolability

- References:
1. ALWR URD, Vol. III, Chapter 5, Section 6.6.3.2, "Cavity Sizing to Promote Long-Term Debris Coolability."
 2. ARSAP Technical Support for the EPRI Debris Coolability Requirement for Advanced Light Water Reactors.

AP600 Position:

The AP600 containment is arranged so that core debris can spread over an area not less than 0.02 square meters per megawatt rated thermal power. The requirement ensures that the decay heat can be removed from the debris and that safe stable state is achievable. The requirement is based on experimental and analytical results that show the debris will spread and form a continuous layer that is coolable by the ingression of water. If the debris is in a configuration that is not coolable, given room, it will spread into a coolable configuration. The debris surface will crack due to shrinkage and allow the ingression of water. The amount of heat that can be removed from the debris is limited by the hydrodynamic stability of the counter-current water steam flow within the debris. The 0.02 square meters per megawatt thermal power is based on a conservative estimate of this heat flux and a conservative factor of safety at a decay heat level of one percent. Chemical energy due to oxidation reactions is not considered since this source of energy is limited.

The current AP600 containment design employs a cavity with a total surface area of 46.5 square meters. The AP600 core has a rated thermal power of 1933 MWt.

This yields a coolability ratio of 0.024 square meters per MWt, well above the requirement. Additionally, in an severe accident situation, the AP600 design ensures that water is available in the cavity for cooling debris at the time of vessel failure. In a LOCA situation, the water is delivered from the break in the RCS. In a loss of heat sink severe accident the water comes from IRWST overflow. After vessel failure, the IRWST discharge into the vessel downcomer will deliver the water to the cavity through the hole in the vessel. The decay heat is ultimately removed from the containment via the Passive Containment Cooling System and the condensate is returned to the cavity.

Any increase in the spreading area beyond 0.024 square meters per MWt will necessitate a major redesign effort on the containment and layout. A significantly large number may require a containment design that is not economically viable.

Need date to Support AP600 Design: Now

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.2.3 Seismic Hazard

Reference: ALWR URD, Chapter 1, Appendix A, 3.3

AP600 Position:

The application of a seismic hazards PRA versus a seismic margins evaluation is currently unresolved. Westinghouse is working with EPRI and the NRC to resolve the issue, and will utilize the approach agreed to be acceptable.

Need Date to Support AP600 Design: NOW

10.2.4 Hydrogen Control

Reference: ALWR URD, Vol. III, Chapter 1, Appendix B, Section 2.5.3.1

AP600 Position:

The AP600 is designed to provide control of hydrogen to ensure that hydrogen concentration limits are not exceeded in reasonably postulated core damage events. Both containment size and combustible gas control systems are utilized to meet this requirement. The AP600 is capable of accommodating hydrogen generation resulting from oxidation of 75 percent of the active fuel cladding with a uniform hydrogen concentration less than 13 percent on a dry volume basis. The potential effects of hydrogen burning will be evaluated and shown to be acceptable.

The definition of the percent active cladding which oxidizes and the uniform hydrogen concentration warrants continued consideration for the AP600, because the passive design differs from earlier Evolutionary designs. Automatic depressurization and early cavity flooding limit the generation of hydrogen. The larger containment to thermal power ratio reduces the average hydrogen concentration. The presence of steam tends to inhibit detonation.

Westinghouse is working with the ALWR hydrogen working group to determine the acceptability of these criteria to the NRC Staff. Final design of the hydrogen control systems will be based on the outcome of these discussions.

Need Date to Support AP600 Design: 9/91

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.3 Source Term Treatment

10.3.1 Use of Physically-Based Source Term

Reference: ALWR URD, Vol. III, Chapter 1, Appendix B, Section 2.5.2, "Source Term for the Passive ALWR"

AP600 Position:

The AP600 will use a physically-based source term as the basis for accident mitigation system design and regulatory dose analyses. The physically-based source term is based on core damage events chosen to provide a conservative estimate of the source term.

The criteria used to estimate the source term involve core damage at low RCS pressure with vessel melt-through, realistic timing of fission product release, realistic fission product release magnitude and chemical form, and realistic aerosol removal coupled to containment conditions for the sequences. The source term estimate utilizes the body of source term knowledge which has been developed over the past decade and for which there is general industry-NRC agreement.

The differences between the mechanistic and non-mechanistic source terms in areas such as timing and chemical composition make the design of mitigation features that would cover both cases difficult. The initiation and duration of a containment spray system and the time required for containment isolation valves to close are two examples.

AP600 design features are being optimized assuming the EPRI mechanistic approach to a physically based source term will be accepted. If the EPRI source term is not adopted, the resultant AP600 design will not take advantage of the knowledge gained on source term behavior and will result in system designs that are less than optimum.

Need Date to Support AP600 Design: NOW

10.3.2 Radionuclide Attenuation

Reference: ALWR URD, Vol. III, Chapter 5, Section 6.4.3.1 and 6.4.3.5

AP600 Position:

Westinghouse is working with the ALWR working group to determine what methods are best suited to provide this function in the Passive Pressurized ALWR. This approach will be submitted to the NRC for review and approval.

Need Date to Support AP600 Design: 8/91

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.4 Emergency Planning

10.4.1 Off-site Emergency Planning

Reference: ALWR URD, Vol. III, Chapter 1, Appendix B, Section 2.1.4.1,
"Off-site Emergency Planning"

ALWR Position:

The basis for the simplification of off-site emergency planning for the AP600 is the plant design requirements which provide very low probability of core damage and, in the event of a core damage accident, very high assurance of containment integrity and low off-site dose.

10CFR 50.33 and 10CFR 50.47 state that licensing applications must have an emergency plan and suggest a plume exposure pathway emergency planning zone (EPZ) of about 10 miles and an ingestion pathway EPZ of about 50 miles. EPZs are discussed in NUREG-0396, which recommends the 10-mile plume exposure pathway EPZ on the basis that this size provides:

1. Dose savings to the population in areas where the projected dose from DBAs could exceed the Protective Action Guides (PAGs) under unfavorable atmospheric conditions,
2. Consequences of less severe Class 9 accidents which would not exceed the PAGs inside the EPZ, and
3. Substantial reduction of early severe health effects in the event of more severe Class 9 accidents.

The actual determinant of the 10-mile limit is based upon the third reason, since the probability of large doses (>200 rem) drops off substantially beyond about 10 miles.

The Protective Action Guides, established by the Environmental Protection Agency, set the limits for exposure to airborne radioactive material following an incident. Off-site emergency planning is based upon initiating protective actions, to limit the exposure to the public, at the exposure levels listed in the PAGs.

The Westinghouse position is that the emergency plan regulations applied to the AP600 can be changed to significantly simplify off-site emergency planning. The basis for the simplification is that the plant design provides a very low probability of core damage and, in the event of a core damage accident, very high assurance of containment integrity and low off-site dose.

AP600 CERTIFICATION PROGRAM DESCRIPTION

The simplification would primarily be in eliminating the need for certain plume exposure pathway related requirements including early notification of the public, dissemination of emergency planning information to the public, analysis of the time required to evacuate and for taking other protective actions for various sectors and distances, and provisions for exercising the off-site plan. The simplification would also eliminate the need for the capital equipment for these actions. The on-site emergency plan and a limited number of plume exposure pathway planning and ingestion pathway elements would be retained to provide a base of expansion of response efforts in the event such expansion is necessary.

Westinghouse is developing design features that can be applied to the AP600 to achieve meeting the PAG exposure limits at 1/2 mile.

Need Date to Support AP600 Design: Now

10.4.2 Design Basis Duration for Safety Systems

Reference: ALWR URD, Vol. III, Chapter 5, Section 1.2.1, "Passive Safety Systems"

AP600 Position:

The AP600 was designed with reliance on passive systems. The passive systems are designed and sized to perform their safety function independent of operator action or off-site support for 72 hours after the initiating design basis event. The objective of the AP600 design is to use passive systems to replace all the active engineered safety systems presently used in light water reactors. A primary benefit of this design philosophy is to provide very simple and extremely reliable safety systems that do not require a complex network of support systems, including safety-grade ac power and cooling water systems that rely on pumped fluids, while meeting the NRC safety goals and requirements.

The general AP600 approach is to provide indefinite core and containment cooling with the passive safety systems even after 72 hours from the initiation of a design basis accident. Other safety functions are permitted to make use of temporary equipment after 72 hours as long as reasonable supplies are readily available. Connections for the temporary equipment will be designed into the plant.

In addition to installed non-safety systems, provisions have been designed into the AP600 to provide the following functions:

- o Replenishing the Spent Fuel Pool water to prevent uncovering the fuel assemblies
- o Refilling the Passive Containment Cooling tank to maintain containment cooling. Failure to refill the tank results in increased containment temperatures and pressures, but the containment will remain below its design pressure

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- o Recharging the batteries through a small temporary battery charger to maintain the instrumentation required to monitor plant status
- o Rejuvenating the control room air supply to maintain air quality and positive pressure in the control room

Need Date to Support AP600 Design: NOW

10.5 Emergency Shutdown

Reference: ALWR URD, Vol. III, Chapter 5, Section 3.3.2

ALWR URD, Vol. III, Chapter 1, Appendix B, Section 2.5.6, "Safe Shutdown"

AP600 Position:

The AP600 will employ safety-grade decay heat removal systems which will enhance the capability, relative to current plants, of maintaining the plant in a safe shutdown condition following non-LOCA events. It is not necessary that these passive safety systems be capable of achieving cold shutdown (i.e., 200°F for PWRs) as required by Regulatory Guide 1.139.

GDC 34 requires the capability of removing decay heat after shutdown so that acceptable design limits of the fuel and reactor coolant pressure boundary are not exceeded. Originally, the capability to maintain PWRs at hot standby was considered sufficient to meet this requirement. Over time, however, it was recognized that to ensure long-term cooling with the typical complement of active systems, the plant must eventually be cooled down and depressurized to the point where low pressure RHR can be employed. In 1978, Regulatory Guide 1.139 was issued requiring the capability to achieve cold shutdown in 36 hours using only safety grade systems.

The AP600 design is based upon meeting the requirements of GDC 34 in an improved manner. In particular, the passive RHR system is employed for both hot standby and long-term cooling modes. This system can operate at full RCS pressure, so that only one system is necessary to place the RCS in the long-term cooling mode. Hot standby conditions are achieved immediately and a temperature of 400°F can be reached within 72 hours. Reactor pressure can be reduced to about 250 psig if desired.

Since the passive RHR system can maintain safe conditions indefinitely, cold shutdown is only necessary to gain access to the reactor coolant system for inspection or repair. On the AP600, this will be accomplished by the non-safety reactor shutdown cooling systems (e.g., startup feedwater, normal RHR, component cooling, service water, ultimate heat sink). These non-safety systems are highly reliable; they have the same redundancy as current generation safety-grade systems and can be supplied with AC power from either on-site or off-site sources. Because these systems are used for all normal cooldowns, there is high assurance they will be available when called upon.

Need Date to Support AP600 Design: Now

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.6 Seismic Engineering

10.6.1 Elimination of Operating Basis Earthquake

Reference: ALWR URD, Vol. III, Chapter 1, Appendix B, Section 2.1.1,
"Operating Basis Earthquake and Dynamic Analysis Methods"

AP600 Position:

The Operating Basis Earthquake has been eliminated as a design requirement for safety related structures and systems. The AP600 design is based on a single earthquake.

For safety related items, the design earthquake is the Safe Shutdown Earthquake. In specifying design criteria for this earthquake, consideration is given to lower magnitude earthquakes having a greater probability of occurrence, as well as to larger magnitude earthquakes having a lower probability. For example, cyclic stresses due to earthquakes are included in the design of those items sensitive to fatigue. Analysis methods and allowable stresses assure that there is margin above the SSE.

The ALWR Requirements establish thresholds for actions required of the plant operator after an earthquake has occurred. Recordings from the seismic monitoring system are evaluated against the SSE design basis for the safety related structures and equipment. Thresholds are based on the methodology described in EPRI Reports NP-5930 and NP-6695.

This position includes consideration of earthquakes less than the SSE in the design of the following areas:

- o Design of certain piping systems where the allowable stresses for SSE may be significantly above yield.
- o Qualification testing of equipment so that any effects of a lesser earthquake are introduced prior to the SSE test.

The Standard Safety Analysis Report (SSAR) to be submitted for Design Certification will include results of analyses demonstrating qualification of key AP600 structures and equipment for generic seismic loads. The SSAR will include the Floor Response Spectra to be used in the detail design phase for equipment qualification.

Need Date to Support AP600 Design: Now

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.6.2 Dynamic Analysis Methods

Reference: ALWR URD, Vol. III, Chapter 1, Sections 4.4.3, 4.7.3, and Appendix B, Section 2.1.1

AP600 Position:

The AP600 is being designed for generic seismic conditions for a wide range of sites. This results in very broad band and conservative response spectra. Relaxations in dynamic analysis and design methods will simplify the design for generic sites.

Design criteria for piping systems for the AP600 must be specified in a consistent manner based on a realistic estimate of behavior and capacity. There are a series of issues to be considered and resolved prior to detailed design and analysis. Some of these issues are noted below along with the action requested of the NRC. While all of these issues are applicable to piping, certain of them would also be applicable to other equipment qualification analyses.

- 1) Use of Code Case N411 for time history seismic analysis.

The Code Case damping values have been accepted by the NRC for use in response spectra analysis but not for use in time history analysis. The NRC has accepted 4% SSE damping for time history analysis of the primary loop piping system, and 3% or 2% SSE damping for time history analysis of other piping systems. The Code Case damping should be accepted for all time history analysis.

- 2) Use of Code Case N411 for independent support motion seismic response spectra analysis.

The Code Case damping values have been accepted by the NRC for use in uniform envelop response spectra analysis. The NRC has accepted 4%, 3% or 2% SSE damping for independent support motion analysis of piping systems. The Code Case damping should be accepted for independent support motion analysis.

- 3) Analysis methods for subsystems supported by equipment.

The Standard Review Plan does not provide recommended methods for decoupling "large" equipment from the seismic analysis of "small" supported subsystems. The NRC has accepted decoupling of larger diameter piping from the analysis of smaller diameter piping when the diameter ratio is three or more without the additional requirement of using amplified response spectra at the connection point to the larger diameter piping. Decoupling of "large" equipment from the seismic analysis of "small" supported subsystems should be accepted.

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- 4) Experienced based design guidelines for seismic design of small bore piping.

The NRC has accepted design by analysis for all piping systems. The use of design guidelines for seismic design of small bore piping systems should be accepted.

- 5) Independent support motion response spectra analysis.

The NRC has accepted the independent support motion response spectra analysis when the contributions from each support are combined by absolute sum. When the contributions of each support are combined by square-root-sum-of-the-squares method, this analysis method should be accepted.

- 6) Combination of inertial and seismic anchor motion responses.

The NRC has accepted the absolute sum method for combining inertial and seismic anchor motion responses. Combination by square-root-sum-of-the-squares method should be accepted.

- 7) Combination of modal responses.

The NRC has accepted modal combination methods which give results for closely spaced modes that are between the square-root-sum-of-the-squares combination method and the absolute sum method. Methods which provide for algebraic combination of closely spaced modes should be accepted.

- 8) Response spectra lowering and broadening.

The NRC recommends that response spectra be broadened 15% to account for modeling uncertainties. Lowering and broadening recommended in ASCE Standard 4-86 should be accepted.

- 9) Response spectra shifting.

Response spectra shifting considers that only one frequency in a group of frequencies can be on the peak of the spectra. The response spectra shifting recommended in ASME III appendix N should be accepted.

- 10) Load combinations for seismic category components.

The NRC recommends combining SSE and pipe rupture loads. The load combinations recommended by the ALWR document which does not combine SSE and pipe rupture should be approved.

- 11) Mechanical loads for postulated plant events.

The NRC recommends that mechanical loads that result from all postulated plant events should be evaluated to ASME limits for seismic category components. Certain unanticipated plant events are only postulated to provide margin for internal pressure and fatigue life (e.g. locked pump rotor). A separate list of plant events for design mechanical loads should be approved.

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12) Stress criteria for secondary loads.

There are no ASME or NRC requirements for certain secondary stress producing loads (e.g. SSE seismic anchor motion, emergency and faulted condition thermal expansion). Interim criteria should be approved pending changes to the ASME Code.

13) Allowable piping stresses for SSE.

The current ASME Code treats SSE as a primary load with a stress limit of 2.0Sy or 3.0S. Code revisions are being discussed to increase the stress limits or reclassify SSE as a secondary load. Interim criteria for SSE should be approved, pending changes to the ASME Code.

Piping design criteria are currently being established for the AP600. It is Westinghouse's intent to develop a standard design suitable for a variety of soil and rock sites. It is important not to include excessive conservatism in the design criteria in order to minimize the need for a large number of supports and snubbers.

Need Date to Support AP600 Design: 7/1/91

10.7 Advanced Digital Control Systems

Reference: ALWR URD, Vol. III, Chapter 10, "Man-Machine Interface Systems"

AP600 Position:

The AP600 instrumentation and controls have been integrated into overall systems, control systems, etc., so that a fully satisfactory interface is achieved between the plant staff and the plant systems and equipment.

The M-MIS employs modern digital technology in a distributed architecture to implement the monitoring, control and protection functions for the AP600. Where it is appropriate, multiplexed data communications are used for any function, including safety functions, to reduce the cost and complexity of the instrumentation and control cable runs throughout the plant. Where appropriate, a fiber optic medium for data transmission is used to provide high data transmission rates with electrical isolation and protection from electromagnetic interference.

The traditional main control board with dedicated meters, recorders and control switches is replaced in the AP600 design by work stations that provide for the electronic display of:

- o Normal, abnormal and emergency operating procedures
- o Plant operational parameters and technical data based on operator tasks and event categories in graphical and diagrammatic format
- o Piping and instrument diagrams generated from computer aided design software
- o Alarms designed specifically to minimize nuisance alarms and the control of devices and systems by a combination of soft control devices and some dedicated control switches

AP600 CERTIFICATION PROGRAM DESCRIPTION

Significant improvements in system testing are provided in the AP600 M-MIS design. The equipment is designed and configured to readily support in-service testing by incorporation of good human factors principles, avoiding the use of undesirable features such as the addition of jumpers or lifting leads, and providing built-in test features, including self diagnostics for continuous on-line testing and automated functional testing for periodic surveillance testing.

Most M-MIS equipment is located in compartments with controlled environments, maintained by reliable HVAC systems. All M-MIS equipment will be selected to be compatible with its environment under normal and casualty conditions as appropriate to meet functional requirements.

The display, control, and protection features are designed to satisfy existing regulations, for example: separation and independence requirements for Class 1E circuits (IEEE Standard 384); criteria for protection systems (IEEE Standard 279); and requirements for manual initiation of protective actions at the system level (Regulatory Guide 1.62).

The AP600 M-MIS will meet existing regulatory requirements. It is noted, however, that many existing regulatory guides and IEEE standards applicable to man-machine interface systems and equipment do not specifically account for the potential use of modern digital technology. Thus, evolution of the detailed M-MIS design may necessitate new interpretation of the intent and purpose of existing guides and standards.

Need Date to Support AP600 Design: Now

10.8 Leak-Before-Break for Subcompartment Design

References: ALWR URD, Vol III. CHAPTER 5, SECTION 8.2.2.4, "Subcompartment Pressure Capability"

ALWR URD, Vol III. CHAPTER 1, SECTION 4.5.5.1 "In-plant Hazards, Rupture of Piping"

ALWR Position:

"The ALWR is required by 10CFR50, Appendix A, to withstand an arbitrary break of any size up to and including a double ended guillotine break of any pipe inside containment as the design basis for the containment structure and for the ECCS. For this deterministic event, dynamic effects on structures, equipment, and components resulting from this event shall not be explicitly considered in the design. These events include pipe whip, jet impingement, rapid subcompartment pressurization, hydraulic system internal loads, and the motion of equipment attached to components responding to these effects."

"For pipe ruptures whose dynamic effects are eliminated by leak-before-break (LBB), static and environmental effects constitute the design bases for plant structures, systems and equipment. The static effects of postulated pipe rupture include subcompartment pressurization, flooding, and thermal expansion of equipment or structures wetted by hot fluids. Environmental effects include flooding, humidity, temperature, exposure to corrosive fluids, and potentially radioactive fluids."

AP600 CERTIFICATION PROGRAM DESCRIPTION

The AP600 is being designed to the ALWR requirements. The static effects of subcompartment pressurization are based on ANSI/ANS 58.2 which requires evaluation of subcompartment pressurization for the leakage crack. This approach has been accepted by the NRC staff for existing plants.

The position is consistent with the Westinghouse comments on the draft Standard Review Plan 3.6.3 published in the Federal Register on August 28, 1987. Comments related to subcompartment pressurization were as follows:

Delete section V, Item 2 of SRP 3.6.3 (Draft) and replace by the following:

"Evaluation of structures protecting essential equipment is still required. Design of these structures includes all dynamic effects of high energy piping not qualifying for leak-before-break. (See B.1.c(4) of MEB 3-1). When LBB is used to eliminate dynamic effects, pressurization by alternative sources of leakage in the compartment should be reviewed, such as flanges, valves and bolted covers."

Explanation

There is no technical basis for imposing a single area leakage with 3-second linear release time as stated in the draft of SRP 3.6.3. If this large leakage area did exist, the pipe would experience complete severance due to internal pressure loads. Leak-before-break evaluations demonstrate that a much smaller leakage (approximately 10.0 gpm) will be stable and detectable with margin.

The elimination of the subcompartment pressurization loading, when LBB is demonstrated, provides the following substantial benefits:

- a. Concrete structures such as the reactor primary shield wall or main steam tunnel outside containment can be more effectively designed to perform their required functions, e.g., radiation shielding and protection from external missiles.
- b. Steel structures such as the primary equipment supports or blowout panels outside containment can be more effectively designed for thermal expansion, seismic and external missiles.

Need Date to Support AP600 Design: Now

AP600 CERTIFICATION PROGRAM DESCRIPTION

10.9 Tornado Design

References: ALWR URD, Vol. III, Chapter 1, Table 1.2-6 and Appendix B, Section 2.1.3

ANSI/ANS-2.3-1983, Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Plant Sites

AP600 Position:

The AP600 design for tornado effects will be according to the Design Basis Tornado characteristics of ANSI/ANS 2.3-1983 for an annual probability of occurrence of $1.0E-6$ and tornado missile Spectrum I of SRP 3.5.1.4. Also, the ALWR will eliminate certain missile provisions, such as labyrinths on HVAC inlet and exhaust.

Regulatory Guide 1.76 establishes tornado design parameters. This Regulatory Guide is referenced from SRP 2.3.1. ANS Standard 2.3 provides an alternate approach. Westinghouse proposes to utilize the set of tornado parameters given in ANS 2.3. The tornado parameters are also used to establish tornado missile velocities in SRP 3.5.1.4 and minimum concrete thicknesses for exterior walls in SRP 3.5.3.

The Design Basis Tornado for the AP600 is defined by the following parameters (ANS 2.3 terminology):

- Maximum wind speed - 260 mph (rotational plus translational speed)
- Translational speed - Maximum 57 mph, minimum 5 mph
- Radius to maximum wind from center of tornado - 453 ft.
- Atmospheric pressure drop - 1.46 psi

The tornado missiles are defined in accordance with SRP 3.5.1.4, and barrier thicknesses in accordance with SRP 3.5.3, with the velocities adjusted to the maximum wind velocity defined above. The following missiles are postulated:

- a) a massive high kinetic energy missile which deforms on impact, assumed to be a 4000 lb automobile impacting the structure at normal incidence with a horizontal velocity of 91 mph or a vertical velocity of 64 mph. This missile shall be considered at all plant elevations up to 30 feet above grade.
- b) a rigid missile of a size sufficient to test penetration resistance, assumed to be a 275 pound 8" armor piercing artillery shell impacting the structure at normal incidence with a horizontal velocity of 91 mph or a vertical velocity of 64 mph. Exterior concrete walls with a minimum thickness of 12 inches and strength of 4000 psi and concrete slabs with a minimum thickness of 9 inches and strength of 4000 psi provide acceptable protection against this postulated missile.

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- c) a small rigid missile of a size sufficient to just pass through any openings in protective barriers, assumed to be a 1" dia. solid steel sphere assumed to impinge upon barrier openings in the most damaging direction at a velocity of 91 mph.

Openings through the exterior walls of the Nuclear Island, and the location of equipment in the vicinity of such openings, are arranged to limit the probability that a missile would prevent safe shutdown of the plant.

Need date to Support AP600 Design: Now

10.10 Industry Codes and Standards

Reference: ALWR URD, Vol. III, Chapter 1, Table 1.4-3

AP600 Position:

Many existing NRC criteria are based on outdated codes and standards and need to be updated. In some instances, the more recent codes and standards are less restrictive than past criteria and NRC need to establish a position on each code. This issue is applicable to all disciplines.

Design of the AP600 plant systems and structures has started based on the criteria incorporated in the ALWR URD. Westinghouse is preparing a list of specific codes and standards that are being applied. These lists will be transmitted to the NRC and discussions on individual areas will be initiated in June, 1991. NRC judgement on the application of these codes will be requested.

Need Date to Support AP600 Design: 11/1/91

10.11 In-Service Testing of Pumps and Valves

References: ALWR URD, Vol. III, Chapter 1, Section 8.6,
"Inspection and Testing"

SECY-90-016, Evolutionary Light Water Reactor Certification Issues
and Their Relationships to Current Regulatory Requirements

AP600 Position:

Components, including pumps and valves classified as ASME Section III, Class 1, 2, or 3, and the function of which is safety related, will be designed and be provided with access to enable the performance of in-service inspection and testing for assessing operational readiness as set forth in the applicable edition and addenda to Section XI of the ASME Boiler and Pressure Vessel Code and Addenda.

AP600 CERTIFICATION PROGRAM DESCRIPTION

Recommended ISI techniques will be the same as techniques used to inspect equipment during manufacture. The equipment procurement specification will either provide the detailed procedures to be used both in the manufacture and at the site or will require that they be provided by the equipment vendor. The AP600 RAM program will define an appropriate frequency for the disassembly and inspection of pumps and valves.

The AP600 design will permit performing as many surveillance tests as possible at power without interfering with normal operation, lifting loads, or blocking relays physically. Provisions will permit testing systems/subsystems or instrument loops in as close to normal operating conditions as practical, although in the case of some remote valves, such as the ADS valves, the test conditions can not be very close to actual operation.

Surveillance testing will utilize non-intrusive techniques or techniques that minimize the amount of wear caused by the surveillance tests themselves. Provisions are included for on line leak testing of the check valves and for forward flow tests.

The AP600 plant is being designed to support In Service Testing of components as required by ASME Code and regulations. In addition, designers are actively doing studies for evaluation and incorporation of on-line diagnostic and monitoring systems consistent with identified testing and monitoring requirements. These requirements are being added to the equipment specifications for inclusion into the designs. Westinghouse believes that the approach we are taking for the AP600 will satisfy the issues the Staff has raised for the inspections and operability of pumps and valves.

Need Date to Support AP600 Design: Now