1.9 Compliance with Regulatory Criteria

1.9.1 Regulatory Guides

Regulatory guides are issued by the NRC in the following 10 broad divisions:

- Division 1 - Power Reactors
- Division 2 - Research and Test Reactors
- Division 3 - Fuels and Materials Facilities
- Division 4 - Environmental and Siting
- Division 5 - Materials and Plant Protection
- Division 6 - Products
- Division 7 - Transportation
- Division 8 - Occupational Health
- Division 9 - Antitrust and Financial Review
- Division 10 - General

Divisions 2, 3, 6, 7, 9, and 10 of the regulatory guides do not apply to the design and design certification phase of AP1000. The following sections provide a summary discussion of NRC Divisions 1, 4, 5, and 8 of the regulatory guides applicable to the design and design certification phase of AP1000.

Appendix 1A provides a discussion of AP1000 regulatory guide conformance.

1.9.1.1 Division 1 Regulatory Guides - Power Reactors

Currently there are approximately 190 Division 1 regulatory guides that have been issued by the NRC for implementation or for comment.

Appendix 1A provides an evaluation of the degree of AP1000 compliance with NRC Division 1 regulatory guides. The revisions of the regulatory guides against which AP1000 is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. For those regulatory guides applicable to the AP1000 Table 1.9-1 identifies the appropriate DCD cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

The superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

1.9.1.2 Division 4 Regulatory Guides - Environmental and Siting

One Division 4 regulatory guide, Regulatory Guide 4.7, merits discussion.

Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," provides guidelines for identifying suitable candidate sites for nuclear power stations. The guidance of this regulatory guide is considered as appropriate in the establishment of the AP1000 site interface criteria, and is described in Sections 2.1 and 2.5.

1.9.1.3 Division 5 Regulatory Guides - Materials and Plant Protection

Three Division 5 regulatory guides, Regulatory Guides 5.9, 5.12, and 5.65, merit discussion.

Regulatory Guide 5.9, "Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material," provides guidelines for data acquisition systems associated with the use of a lithium-drifted germanium gamma ray spectroscopy system. This regulatory guide is not applicable to AP1000 design certification.

Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," provides guidelines for the selection and use of commercially available locks in the protection of facilities and special nuclear material. The guidance of this regulatory guide is considered as appropriate in the AP1000 design.

Regulatory Guide 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," is not applicable to design certification.

1.9.1.4 Division 8 Regulatory Guides - Occupational Health

Two Division 8 regulatory guides, Regulatory Guides 8.8 and 8.19 merit discussion.

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA)," provides NRC guidance for meeting the requirements of 10 CFR Part 20. This regulatory guide includes guidance in the following areas for maintaining radiation exposures ALARA:

- Overall program (e.g., policy, organization, and training)
- Facility and equipment design features
- Radiation protection program
- Radiation protection facilities, instrumentation, and equipment

Regulatory Guide 8.8 is written primarily for utility applicants and licensees. However, Westinghouse has established policy, design, and operational considerations that will be applied in the AP1000 design in accordance with this regulatory guide. These considerations are discussed in Section 12.1.

Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants" describes a method acceptable to the NRC staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process involved in designing a light-water-cooled power reactor so that occupational radiation exposures will be ALARA. This regulatory guide includes guidance for estimating occupational radiation exposures (principally during the design stage) as a result of:

- Reactor operations and surveillance
- Routine maintenance
- Waste processing
- Refueling

- Inservice inspection
- Special maintenance

Occupational radiation exposure estimates that are in accordance with Regulatory Guide 8.19 are described in Section 12.4.

1.9.1.5 Combined License Information

The Combined License applicant will address conformance with regulatory guides that are not applicable to the certified design or not addressed by the activities required by COL Information Items. The Regulatory Guides included in this Information Item are as follows:

- Regulatory Guide 1.86, Revision 0, 6/74 – Termination of Operating Licenses for Nuclear Reactors

- Regulatory Guide 1.111, Revision 1, 7/77 – Methods for Estimating Atmosphere Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

- Regulatory Guide 1.113, Revision 1, 4/77 – Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I

- Regulatory Guide 1.159, Revision 0, 8/90 – Assuring the Availability of Funds for Decommissioning Nuclear Reactors

- Regulatory Guide 1.162, Revision 0, 2/96 – Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels

- Regulatory Guide 1.174, Revision 0, 7/98 – An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

- Regulatory Guide 1.179, Revision. 0, 9/99 – Standard Format and Content of License Termination Plans for Nuclear Power Reactors

- Regulatory Guide 1.181, Revision 0, 9/99 – Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)

- Regulatory Guide 1.184, Revision 0, 8/00 – Decommissioning of Nuclear Power Reactors

- Regulatory Guide 1.185, Revision 0, 8/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report

- Regulatory Guide 1.186, Revision 0, 12/00 – Guidance and Examples of Identifying 10 CFR 50.2 Design Bases

- Regulatory Guide 1.187, Revision 0, 11/00 – Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments

- Regulatory Guide 5.9 Revision 2, 12/83 – Specifications for Ge (Li) Spectroscopy Systems for Material Protection Measurements Part 1: Data Acquisition Systems

1.9.2 Compliance with Standard Review Plan (NUREG-0800)

WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," provides the results of a review of the AP1000 compliance with the acceptance criteria for each section of the Standard Review Plan, NUREG-0800.

1.9.3 Three Mile Island Issues

This section identifies the Three Mile Island issues of 10 CFR 50.34(f) that are addressed by AP1000 design features or program plans. The additional issues of NUREG-0660 and NUREG-0737 that apply to the AP1000 are resolved in accordance with the guidance of NUREG-0933, with specific details provided in the applicable sections of the DCD.

Some of the 10 CFR 50.34(f) issues initially identified as applicable only to Boiling Water Reactors (BWRs) or Babcock and Wilcox plants have also been addressed for the AP1000 design. For example, the AP1000 design incorporates an automatic depressurization system with some similarity to that utilized for BWRs.

10 CFR 50.34(f):

(1)(i) Plant/Site Specific TMI-Related Risk Assessment (NUREG-0660 Item II.B.8)

"Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

AP1000 Response:

A plant-specific Probabilistic Risk Assessment (PRA) performed on the AP1000 design evaluates the plant in terms of core damage frequency and containment integrity. The PRA supports the design effort and establishes the capability of the design to meet established safety goals. Level 1 (Plant), 2 (Containment), and 3 (Site) PRA evaluations, including internal and external events:

- Demonstrate that the plant design meets the NRC safety goals
- Identify design vulnerabilities, evaluate alternate design features and operational strategies, and modify the design to reduce risk

The PRA process has been integrated into the design process to verify that the design effort meets the targeted goals and resolves the identified vulnerabilities. As a result, specific design changes were incorporated into the plant systems to improve the reliability of the core and containment heat removal systems.

Close interaction between the plant designers and PRA analysts is maintained to consider severe accident vulnerabilities as part of the design process. The AP1000 PRA is provided to the NRC as a separate document.

(1)(ii) Auxiliary Feedwater System Evaluation (NUREG-0737 Item II.E.1)

"Perform an evaluation of the proposed auxiliary feedwater system, to include (applicable to pressurized water reactors only): (A) a simplified Auxiliary Feedwater System reliability analysis using event-tree and fault-tree logic techniques, (B) a design review of Auxiliary Feedwater System, and (C) an evaluation of Auxiliary Feedwater System flow design bases and criteria."

AP1000 Response:

The AP1000 design does not utilize an auxiliary feedwater system. A nonsafety-related startup feedwater system is provided to remove the core decay heat after the reactor trip during postulated non-LOCA event. Decay heat removal maintains core subcooling and prevents water relief from the pressurizer safety valves by preventing heatup of the reactor coolant system. The startup feedwater pumps automatically start following anticipated transients resulting in low steam generator level. However, operation of the nonsafety-related startup feedwater system is not credited to mitigate licensing design basis accidents described in Chapter 15.

The safety-related passive core cooling system provides emergency core decay heat removal during transients, accidents, or whenever the normal nonsafety-related heat removal paths are unavailable.

The safety-related passive core cooling system design basis and criteria are described in Section 6.3.

(1)(iii) Reactor Coolant Pump Seals (NUREG-0737 Items II.K.2.16 and II.K.3.25)

"Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break loss of coolant accident with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss of coolant accident with subsequent reactor coolant pump seal damage."

AP1000 Response:

The AP1000 design uses sealless motor pumps for circulating primary reactor coolant through the reactor core, piping, and steam generators. In the sealless design, all rotating components are inside a pressure vessel; therefore, no seal can fail and initiate reactor coolant system leakage.

(1)(iv) Automatic Power-Operated Relief Valve Isolation System (NUREG-0737 Item II.K.3.2)

"Perform an analysis of the probability of a small-break loss of coolant accident caused by a stuck-open power-operated relief valve. If this probability is a significant contributor to the probability of small-break loss of coolant accidents from all causes, provide a description and evaluation of the effect on small-break loss of coolant accident probability of an automatic power-operated relief valve isolation system that would operate when the reactor coolant system pressure falls after the power-operated relief valve has opened."

AP1000 Response:

The AP1000 design does not include power-operated relief valves. The pressurizer volume is about 40 percent larger than the pressurizer volume in current plants with a comparable power rating. The larger pressurizer increases transient operation margins and prevents safety valve actuation in most accident situations. The pressurizer surge line is also larger to permit a more rapid transfer of coolant between the reactor coolant system and the pressurizer, and also to accommodate the automatic depressurization system first- to third-stage flow rates. The surge line limits the pressure drop during maximum anticipated surge (Condition II loss of load transient) to prevent exceeding the maximum reactor coolant system pressure limit.

Overpressure protection is provided by two totally enclosed pop-type safety valves. These valves are spring-loaded and self-actuated and they are designed to meet the requirements of the ASME Code, Section III. If the pressurizer pressure exceeds the set pressure, the safety valves start lifting. A temperature indicator in the discharge piping for each safety valve alarms on high temperature to alert the operator to the presence of high temperature fluid from leakage or when the valves open.

The AP1000 design also includes an automatic depressurization system. The system consists of four stages of valves. Three stages are connected to the pressurizer. The fourth stage is connected to the hot legs. These valves are not actuated on a high pressure signal. Design features are included to reduce the chance of spurious automatic depressurization system actuation including appropriate interlocks, 2-out-of-4 instrument actuation, fail as is valves, redundant, closed first, second, and third stage valves in each line, and redundant series controllers for forth stage valves. Probabilistic risk assessment is used to determine the probability of a loss of coolant accident caused by failure of the automatic depressurization system. Results of this evaluation are factored into the design process. See Chapter 5 and Section 6.3 for additional information.

(1)(v) Separation of HPCI and RCIC System Initiation Levels (NUREG-0737 Item II.K.3.13)

"Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI')."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(vi) Relief Valve Challenges (NUREG-0737 Item II.K.3.16)

"Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(vii) Automatic Depressurization System Activation (NUREG-0737 Item II.K.3.18)

"Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation to ensure adequate core cooling."

AP1000 Response:

Although this issue is identified as applicable to BWRs only, the AP1000 design uses an automatic depressurization system with some similarity to that used on BWRs.

The automatic depressurization system actuates on Low-1 core makeup tank level, coincident with a core makeup tank actuation signal. Therefore manual actuation of the automatic depressurization system is not required to maintain core cooling. As discussed in Section (1)(i), PRA analysis confirms the reliability of the automatic actuation. Additional information is provided in Section 6.3.

(1)(viii) Core Spray and Low Pressure Coolant Injection Systems (NUREG-0737 Item II.K.3.21)

"Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(ix) RCIC and HPCI Additional Space Cooling (NUREG-0737 Item II.K.3.24)

"Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI')."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(x) Automatic Depressurization System Functionality During Accidents (NUREG-0737 Item II.K.3.28)

"Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves."

AP1000 Response:

Although this issue is identified as applicable to BWRs only, the AP1000 uses a safety-related automatic depressurization system that is different from that presently used on BWRs. The AP1000 automatic depressurization system uses safety-related dc motor-operated valves and squib valves to initiate depressurization. The motive power for these valves is safety-related dc power. There is no nonsafety-related equipment or instrumentation, including instrument air or nitrogen supply, relied on in the operation of these valves.

These valves are designed and qualified to function in the conditions of an accident. They will also be subject of pre-operational and in-service testing. They will be included in the reliability assurance program. Additional information is provided in Section 6.3 for the passive core cooling system, subsection 3.9.3 for valve operability requirements, Chapter 14 for the initial test program, subsection 3.9.6 for in-service testing, and Section 16.2 for the reliability assurance program.

(1)(xi) Depressurization Methods/Rapid Cooldown (NUREG-0737 Item II.K.3.45)

"Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown."

AP1000 Response:

This issue is applicable to BWRs only.

(1)(xii) Hydrogen Control System Evaluation (NUREG-0660 Item II.B.8)

"Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section (50.34). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include: (A) a comparison of costs and benefits of the alternative systems considered, (B) for the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section (50.34), and (C) for the selected system, preliminary design descriptions of equipment, function, and layout."

AP1000 Response:

Continuous indication of hydrogen concentration in the containment atmosphere is provided. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the active zircaloy cladding.

Hydrogen igniters control rapid releases of hydrogen during and after postulated degraded core and core melt accidents to maintain concentration below 10 percent.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

See subsection 6.2.4 for additional information.

(2)(i) Simulator Capability (NUREG-0933 Item I.A.4.2)

"Provide simulator capability that correctly models the control room and includes the capability to simulate small-break loss of coolant accidents."

AP1000 Response:

Simulator capability is not included within the scope of the AP1000 design certification. Functional requirements for simulator capability are derived from Human Factors Engineering Program described in Chapter 18.

(2)(ii) Plant Procedures (NUREG-0933 Item I.C.9)

"Establish a program to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts."

AP1000 Response:

See Chapter 13 of the DCD for a discussion of plant procedures, training of operations personnel and emergency planning.

(2)(iii) Control Room Design (NUREG-0737 Item I.D.1)

"Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts."

AP1000 Response:

The human factors engineering design process of the AP1000 has been developed to conform with NUREG-0711, "Human Factors Engineering Program Review Model." The elements of the design process provide a structured top-down system analysis using accepted human factors engineering principles. The design of the main control room and the other operation and control centers reflect state-of-the-art human factors principles. See Appendix 1A for information on conformance with applicable regulatory guides. See Chapter 18 for additional information on the AP1000 human factors engineering design process.

(2)(iv) Safety Parameter Display System (NUREG-0737 Item I.D.2)

"Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded."

AP1000 Response:

The purpose of the plant safety parameter display console (or safety parameter display system) is to display important plant variables in the main control room in order to assist in rapidly and reliably determining the safety status of the plant.

The requirements for the safety parameter display system are specified during the main control room design process, and are met by the main control room design, specifically as part of the alarms, displays, and controls. The requirements for a safety parameter display system (NUREG-0696, Reference 1) are met by grouping the alarms by plant process or purpose, as directly related to the critical safety functions.

The process data presented on the graphic displays is similarly grouped, facilitating an easy transition for the operators. The safety parameter display system requirement for presentation of plant data in an analog fashion prior to reactor trip is met by the design of the graphic CRT displays.

Displays are available at the operator workstations, the remote shutdown workstation, and at the technical support center. See Chapter 18 for additional information pertaining to the safety parameter display system design.

(2)(v) Safety System Status Indication (NUREG-0933 Item I.D.3)

"Provide for automatic indication of the bypassed and [in]operable status of safety systems."

AP1000 Response:

The AP1000 main control room meets the NRC Regulatory Guide 1.47 recommendations, including automatic indication of bypassed and inoperable status of plant safety systems, as described in Appendix 1A.

Plant safety parameters, protection system status, and plant component status signals are processed by the protection and safety monitoring system and made available to the entire instrumentation and control system via the redundant monitor bus.

Class 1E signals are provided to the qualified data processor, which is part of the protection and safety monitoring system, for accident monitoring displays. The display of this data is incorporated in the process data displays on the graphic CRTs in the AP1000 main control room.

See Chapters 7 and 18 for additional information pertaining to bypass inoperable status indication. Appendix 1A describes conformance with Regulatory Guide 1.47.

(2)(vi) Reactor Coolant System High Point Vents (NUREG-0737 Item II.B.1)

"Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity."

AP1000 Response:

In the AP1000 design, the capability for remotely operated high point venting of the reactor coolant system is provided by the safety-related automatic depressurization system valves and the safety-related reactor vessel head vent system. Both of these vent paths discharge to the in-containment refueling water storage tank.

During loss of cooling accident events, the automatic depressurization system automatically depressurizes the reactor coolant system so that the passive core cooling system may effectively deliver core cooling flow. Depressurization via the automatic depressurization system results in creation of a gas-steam volume in the upper region of the vessel. This vapor volume expands down to the inside of the hot leg before it begins venting through the hot leg either via the automatic depressurization system paths connected to the pressurizer or directly from the hot legs via the fourth stage automatic depressurization system paths. This process provides an open injection and steam venting flow path through the reactor vessel, maintaining required core cooling flow.

The reactor vessel head vent system can also be operated from the main control room to directly vent from the top of the reactor vessel head. Subsection 5.4.12 provides additional information pertaining to the reactor coolant system venting capabilities.

(2)(vii) Plant Radiation Shielding (NUREG-0737 Item II.B.2)

"Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID-14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment."

AP1000 Response:

Post-accident radiation sources, used in the shield design and assessment of post-accident access to vital areas, are addressed in subsection 12.2.1.3. The post-LOCA instantaneous and integrated source strengths as a function of time are also included as Tables 12.2-20 and 12.2-21, respectively. The sources are based on the core activity release model from Regulatory Guide 1.183, which supersedes the TID-14844 source term assumptions as reflected in Regulatory Guide 1.4.

Vital areas for post-accident personnel access are addressed in Section 12.3, including radiation zone maps that show projected dose rates in these areas and access routes for the various post-accident actions in vital areas. Time estimates have been made for ingress, egress, and performance of actions at the vital area locations and have been used in demonstrating that total individual radiation doses are limited to less than 5 rem and that Item II.B.2 of NUREG-0737 and GDC-19 requirements are met.

Environmental qualification of safety-related equipment is addressed in Section 3.11. The determination of the radiation environments during postulated accident situations considers the activity release model based on NUREG-1465, which supersedes the source term definition of Parts 1 and 4 of Item II.B.2 of NUREG-0737.

Subsection 12.2.3 defines the responsibility to address any additional contained radiation sources not identified in 12.2.1. Thus, appropriate source terms have been identified and used in establishing that the requirements of Item II.B.2 of NUREG-0737 and GDC 19 are met and the issues are resolved.

(2)(viii) Post-Accident Sampling (NUREG-0737 Item II.B.3)

"Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 50 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations."

AP1000 Response:

Recently the NRC published a model Safety Evaluation Report on eliminating post-accident sampling system requirements from technical specifications for operating plants (Federal Register Volume 65, Number 211, October 31, 2000). The AP1000 sampling design is consistent with the approach in the Model safety evaluation report and not the guidance outlined in NUREG-0737 and Regulatory Guide 1.97. The primary sampling system design is consistent with contingency plans to obtain and analyze highly radioactive post-accident samples from the reactor coolant system, the containment sump, and the containment atmosphere.

(2)(ix) Hydrogen Control (NUREG-0660 Item II.B.8)

"Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section (50.34) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation."

AP1000 Response:

See the response provided for issue (1)(xii).

(2)(x) Reactor Coolant System Valve Testing (NUREG-0737 Item II.D.1)

"Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for pressurized water reactors, power-operated relief valves, block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed."

AP1000 Response:

The AP1000 reactor coolant system design does not include power-operated relief valves and their associated block valves. However, the safety valve and discharge piping used in the AP1000 design will be either of design similar to those items tested by EPRI and documented in EPRI Report EPRI NP-2770-LD (Reference 2) or will be tested in accordance with the guidelines of Item [II.D.1] of NUREG-0737.

The AP1000 design includes automatic depressurization system valves which are used to depressurize the plant and establish conditions for injection from the accumulators and the in-containment refueling water storage tank. The operability of the automatic depressurization system valves and spargers is confirmed by a test program. See Section 1.5 for information pertaining to the testing program.

Accident analyses for the AP1000 determine fluid conditions expected under operating conditions, transients, and accidents, and the postulated system responses to these conditions, including the operation of reactor coolant system safety valves. Anticipated transients without scram events are analyzed. Appropriate valve qualification documentation is maintained.

(2)(xi) Valve Position Indication (NUREG-0737 Item II.D.3)

"Provide direct indication of relief and safety valve position (open or closed) in the control room."

AP1000 Response:

The AP1000 design does not include power-operated relief valves and their associated block valves from the reactor coolant system.

Direct indication of relief and safety valve position (open or closed) is provided in the main control room.

(2)(xii) Auxiliary Feedwater System Initiation and Indication (NUREG-0737 Item II.E.1.2)

"Provide automatic and manual auxiliary feedwater system initiation, and provide auxiliary feedwater system flow indication in the control room."

AP1000 Response:

As previously noted in the AP1000 response to Issue (1)(ii), the AP1000 design includes a nonsafety-related startup feedwater system, but not an auxiliary feedwater system. Flow indication of the startup feedwater system is provided in the main control room.

The startup feedwater pumps automatically start following anticipated transients resulting in low steam generator level. The startup feedwater control valves automatically control feedwater flow to the steam generators during operation. They can also be operated manually from the main control room.

The safety-related passive core cooling system provides for emergency core decay heat removal during transients, accidents, or whenever the normal heat removal paths are unavailable. Automatic and manual actuation and flow rate indication are available in the main control room.

(2)(xiii) Pressurizer Heater Power Supplies (NUREG-0737 Item II.E.3.1)

"Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available."

AP1000 Response:

The AP1000 pressurizer heaters are powered from the nonsafety-related ac power system. During loss of offsite power events, a portion of the pressurizer heaters is capable of being powered from the nonsafety-related onsite standby power system. The pressurizer heaters are capable of establishing and maintaining natural circulation in hot standby condition, with only the diesel generators supplying electrical power.

With only safety-related dc (Class 1E dc) power available, the safety-related passive core cooling system can establish and maintain natural circulation cooling using the passive residual heat removal heat exchangers, transferring the decay heat to the in-containment refueling water storage tank water and to the passive containment cooling system.

Therefore, the nonsafety-related pressurizer heaters are not required for core decay heat removal following a loss of offsite power. See Section 8.3 for additional information.

(2)(xiv) Containment Isolation System (NUREG-0737 Item II.E.4.2)

"Provide containment isolation systems that: (A) ensure all nonessential systems are isolated automatically by the containment isolation system, (B) for each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs."

AP1000 Response:

The AP1000 containment isolation design satisfies NRC requirements, including post-TMI requirements. In general, this means that two barriers are provided -- one inside containment and the other outside containment. Usually these barriers are valves, but in some cases they are closed, seismic Category I piping systems not connected to the reactor coolant system or to the containment atmosphere. Table 6.2.3-1 identifies containment isolation design provisions for mechanical penetrations. The isolation signal and maximum closure times are defined for each remotely operated valve. Containment penetrations, other than equipment hatches and flanges, incorporate two isolation barriers in series.

The AP1000 design incorporates a reduction in the number of required penetrations compared to the number in previous plant designs. The majority of these penetrations are normally closed. Those few that are normally open, use automatically closed isolation valves.

Containment isolation is automatically actuated by a safeguards actuation signal, using two-out-of-four coincident logic. The containment isolation actuation is set as low as reasonable without creating potential for spurious trips during normal operations. Containment isolation can also be initiated manually from the main control room. Containment penetrations do not automatically reopen on the resetting of the isolation signal. See subsection 6.2.3 for additional information.

(2)(xv) Containment Purging/Venting (NUREG-0933 Item II.E.4.4)

"Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions."

AP1000 Response:

Containment purging for the AP1000 is provided by the nonsafety-related containment air filtration system. The function of the system is to clean up the containment atmosphere to acceptable radiation levels during plant operation and prior to personnel entry. It can also be used for containment pressure equalization.

The containment air filtration system is designed to reliably isolate under accident conditions. There are two penetrations and two containment filtration subsystems for AP1000.

See subsection 9.4.7 for additional information.

(2)(xvi) ECCS Actuation Cycles (NUREG-0933 Item II.E.5.1)

"Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both the expected transients and accidents)."

AP1000 Response:

This issue is applicable to Babcock & Wilcox designs only.

The AP1000 design uses the passive core cooling system to provide emergency reactor coolant inventory control and emergency decay heat removal. Component design criteria have been established for the number of actuation cycles for the passive core cooling system. The identified actuation cycles include inadvertent actuation, as well as the system response to expected plant trip occurrences, including overcooling events.

Automatic depressurization system operation is not expected for either design basis or best estimate overcooling events. See subsection 3.9.1 for additional information.

(2)(xvii) Specific Accident Monitoring Instrumentation (NUREG-0737 Item II.F.1)

"Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples."

AP1000 Response:

AP1000 post-accident monitoring is described in Chapter 7.

AP1000 post-accident monitoring provides for indication of the specified parameters as follows:

- Containment pressure
- Containment water level
- Containment radiation intensity (high level)
- Noble gas effluents - to ascertain reactor coolant system integrity

Other noble gas effluents are designated Type E variables and include information to permit the operators to:

- Monitor the habitability of the main control room

- Monitor plant areas where access may be required to service equipment necessary to monitor or mitigate the consequences of an accident

- Estimate the magnitude of release of radioactive materials through identified pathways

- Monitor radiation levels and radioactivity in the environment surrounding the plant

DCD subsection 11.5.5 has additional information on measurement of radioactive effluents and conformance with Regulatory Guide 1.97.

The AP1000 primary sampling system is designed to provide post accident sampling functions. See DCD subsection 9.3.3.1 for additional information on the post accident sampling system.

The human factors aspects of the AP1000 are discussed in Chapter 18.

(2)(xviii) Inadequate Core Cooling Instrumentation (NUREG-0737 Item II.F.2)

"Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs."

AP1000 Response:

The AP1000 reactor system includes instrumentation for detecting voids in the reactor vessel head and other reactor vessel inventory deficits that could lead to inadequate core cooling.

The available instrumentation includes core subcooling margin monitors, core exit thermocouples, pressurizer level indicators, reactor coolant system reactor vessel level, and reactor coolant pump status (motor current). Reactor vessel level indication is provided from a range in the vessel from the bottom of the hot leg to approximately the reactor vessel mating flange via level instrumentation connected to the hot legs.

The AP1000 features that provide margin to or indication of inadequate core cooling include the following:

- A larger pressurizer than most current PWRs, with a pressurizer that is located above the reactor pressure vessel head
- No automatic power-operated relief valves
- An improved reactor vessel head venting capability
- A passive core cooling system
- A passive containment cooling system
- No dependence on ac power to maintain adequate core and containment cooling
- Reactor coolant system hot leg level instrumentation
- Improved reactor system instrumentation
- Core subcooling monitoring

See Sections 6.3 and 7.5 for additional information.

(2)(xix) Post-Accident Monitoring Instrumentation (NUREG-0933 Item II.F.3)

"Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage."

AP1000 Response:

The AP1000 post-accident monitoring system was developed by using Regulatory Guide 1.97 as a guidance document.

Data used for post-accident monitoring is displayed either by the normal control room display system or by the qualified data processing system.

The normal control room display system is used for display of nonsafety-related signals which are not required to be displayed by a qualified system. The qualified data processing system provides for the display of signals which must be displayed by a qualified system.

The qualified data processing system is a microprocessor-based, safety-related system that provides instrumentation to monitor the plant variables and systems during and following an accident. The system consists of two independent, electrically isolated, physically separated divisions.

Additional details pertaining to this system are provided in the AP1000 response to issue (2)(xvii) and in Chapter 7.

(2)(xx) Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators (NUREG-0737 Item II.G.1)

"Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) level indicators are powered from vital buses, (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources."

AP1000 Response:

The AP1000 design does not include power-operated relief valves and their associated block valves from the reactor coolant system.

Pressurizer level indication is provided by instrumentation powered from the Class 1E dc and UPS system. The system provides safety-related, uninterruptible power for the Class 1E plant instrumentation, control, monitoring, and other vital functions, including safety-related components that are essential for safe shutdown of the plant.

The Class 1E direct current system is designed such that these critical plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are unavailable.

See Chapter 7 and Section 8.3 for additional information.

(2)(xxi) Auxiliary Heat Removal Systems (NUREG-0933 Item II.K.1.22)

"Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable."

AP1000 Response:

Although this issue is applicable to BWRs only, there are some considerations for AP1000.

Following a loss of main feedwater for the AP1000, there are a number of plant systems that automatically actuate to provide decay heat removal. The startup feedwater system is a nonsafety-related system, that can be powered by the nonsafety-related diesel generators, and is automatically actuated and controlled by steam generator level.

For design basis events, the safety-related passive core cooling system includes a passive residual heat removal heat exchanger which automatically actuates to provide emergency core decay heat removal if the nonsafety-related systems are not available.

The AP1000 main control room meets the NRC guidelines for manual actuation of protective functions including those that are used in the event of a loss of normal feedwater.

See Sections 6.3 and 10.4 for additional information.

(2)(xxii) Failure Mode and Effects Analysis for Control Systems (NUREG-0933 Item II.K.2.9)

"Provide a failure modes and effects analysis of the integrated control system to include consideration of failures and effects of input and output signals to the integrated control system."

AP1000 Response:

This issue is applicable to Babcock & Wilcox plants only.

(2)(xxiii) Safety-Grade Anticipatory Reactor Trip (NUREG-0737 Item II.K.2.10)

"Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip."

AP1000 Response:

This issue is applicable to Babcock & Wilcox plants only.

The AP1000 trip logic includes an anticipatory reactor trip for loss of main feedwater using low steam generator water level. See Section 7.2 for additional information.

Since the AP1000 design does not include power-operated relief valves and their associated block valves in the reactor coolant system, the anticipatory reactor trip on turbine trip is not required for AP1000.

(2)(xxiv) Central Water Level Recording (NUREG-0933 Item II.K.3.23)

"Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements."

AP1000 Response:

This issue is applicable to BWRs only.

(2)(xxv) Emergency Response Facilities (NUREG-0737 Item III.A.1.2)

"Provide an onsite technical support center, an onsite operational support center, and, for construction permit applications only, a nearsite emergency operations facility."

AP1000 Response:

The AP1000 provides for an onsite technical support center and an operational support center. See the figures in Section 1.2 for additional information on the location. The detailed design of the workstations and the associated man-machine interface for the technical support center and the operational support center is guided by the human factors engineering design process described in Chapter 18 of the DCD. The offsite emergency response facility is discussed in subsection 18.2.6.

(2)(xxvi) Leakage Control Outside Containment (NUREG-0737 Item III.D.1.1)

"Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency."

AP1000 Response:

As described in issue (2)(vii), the safety-related AP1000 passive systems do not recirculate radioactive fluids outside of containment following an accident. A nonsafety-related system can be used to recirculate coolant outside of containment following an accident, but this system is not operated when high containment radiation levels exist.

(2)(xxvii) In-Plant Monitoring (NUREG-0737 Item III.D.3.3)

"Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

AP1000 Response:

Area radiation monitors (ARMs) are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 1.97, 8.2, and 8.8. In addition to the installed detectors, periodic plant environmental surveillance is established.

(2)(xxviii) Control Room Habitability (NUREG-0737 Item III.D.3.4)

"Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID-14844 source term release, and make necessary design provisions to preclude such problems."

AP1000 Response:

Normally, a nonsafety-related HVAC system keeps the AP1000 main control room slightly pressurized to prevent infiltration of air from other plant areas. During accident conditions, a safety-related isolation of the main control room is automatically actuated.

Upon the loss of nonsafety-related ac power, the main control room environment is sufficient to protect the operators and support the man-machine interfaces necessary to establish and maintain safe shutdown conditions for the plant following postulated design basis accident conditions. The sources are based on the core activity release model from Regulatory Guide 1.183, which supersedes the TID-14844 source term assumptions as reflected in Regulatory Guide 1.4.

The main control room is sealed with safety-related connections to a safety-related compressed air breathing source. This compressed air system provides continued pressurization and a source of fresh air for operator habitability. The air supply is sized to last for 72 hours following an accident. It is expected that the onsite nonsafety-related normal HVAC system will be operational before the installed compressed air supply is exhausted.

The nonsafety-related HVAC system, equipped with a refrigeration-type air conditioning unit, normally provides main control room cooling. This equipment is powered from the onsite diesel generators. If the normal HVAC system is not available, outside air is not allowed into the main control room, and the safety-related compressed air storage system is actuated.

(3)(i) Industry Experience (NUREG-0737 Item I.C.5)

"Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant."

AP1000 Response:

AP1000 design engineers are continually involved in reviewing industry experiences from sources such as NRC Bulletins, Licensee Event Reports, NRC request for information letters to holders of operating licenses for nuclear power reactors, Federal Register information, and generic letters. Lessons learned experience was incorporated in the AP600 through the Westinghouse participation in developing Volume III of the ALWR Utility Requirements Document and participation in the ALWR Utility Steering Committee activities. The AP1000 design is closely based on the AP600. See Section 1.9.5.5 for additional information.

(3)(ii) Quality Assurance List (NUREG-0933 Item I.F.1)

"Ensure that the quality assurance list required by Criterion II, Appendix B, 10 CFR Part 50 includes all structures, systems and components important to safety."

AP1000 Response:

The AP1000 Quality Assurance Plan is described in Chapter 17. Structures, systems, and components are classified as described in Section 3.2.

(3)(iii) Quality Assurance Program (NUREG-0737 Item I.F.2)

"Establish a quality assurance program based on consideration of: (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including Quality Assurance personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining Quality Assurance programmatic requirements; (E) establishing qualification requirements for Quality Assurance and Quality Control personnel; (F) sizing the Quality Assurance staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a Quality Assurance role in design and analysis activities."

AP1000 Response:

The AP1000 Quality Assurance Plan described in Chapter 17 meets the requirements of issue 1.F.2.

(3)(iv) Dedicated Containment Penetrations (NUREG-0660 Item II.B.8)

"Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system."

AP1000 Response:

The containment analysis for the AP1000, including PRA and severe accident assessments, demonstrate that the containment, with its passive heat rejection capability, does not need a filtered vent to prevent overpressurization.

The 36-inch diameter containment air filtration system penetration provided for AP1000 meets the requirement of 10 CFR 50.34(f)(3)(iv). See Figure 9.4.7-1, note 6, for additional information.

(3)(v) Containment Design (NUREG-0660 Item II.B.8)

"Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above, appropriate for each type of containment, will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraph (f)(3)(v)(A)(1) and (f)(3) (v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. . . .

(B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category), (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting."

AP1000 Response:

The AP1000 containment vessel is designed to meet the requirements of the ASME Code, Section III, Division I, Subsection NE. A severe accident containment analysis is conducted to support the design effort. The results of the analysis are fission product source terms and plant thermal-hydraulic response for each of the accident sequences chosen to be representative of the plant damage states determined in level 1 PRA analysis.

Results of the analysis indicate that containment failure is not predicted for cases in which the passive containment cooling system cooling water is available. The hydrogen igniter system controls hydrogen and mitigates threats to the containment due to hydrogen.

See Section 6.2 for additional information.

(3)(vi) Hydrogen Recombiners (NUREG-0737 Item II.E.4.1)

"For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere."

AP1000 Response:

Since external hydrogen recombiners are not provided for the AP1000, this requirement is not applicable. See Section 6.2 for additional information.

(3)(vii) Management Plan (NUREG-0933 Item II.J.3.1)

"Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort."

AP1000 Response:

The AP1000 design team has developed a management plan for the AP1000 project which consists of a properly structured organization with open lines of communication, clearly defined responsibilities, well-coordinated technical efforts, and appropriate control channels. The procedures to be used in the construction, startup, and operation phases of the plant are provided in accordance with the Master Plan and Procedure Development Process identified in APP-GW-GLR-040 (Reference 72).

1.9.4 Unresolved Safety Issues and Generic Safety Issues

Proposed technical resolutions of Unresolved Safety Issues and medium- and high-priority Generic Safety Issues, as identified in NUREG-0933, Reference 3 are required for new plants as part of the NRC policy on severe accidents and are required for design certification in accordance with 10 CFR 52.47(a)(1)(iv).

The current program for identifying and establishing the priority of open safety issues is summarized in NUREG-0933. This program provides for the prioritization and tracking of previously categorized Unresolved Safety Issues and Generic Safety Issues, New Generic Issues, TMI Action Plan Items Under Development, and Human Factors Program Plan Issues.

The following subsection reviews each of the NUREG-0933 safety issues and identifies the safety issues that are applicable to the AP1000. For each of these issues guidance is provided on how the issue is addressed for the AP1000.

1.9.4.1 Review of NRC List of Unresolved Safety Issues and Generic Safety Issues

Applicants for design certification are required by 10 CFR 52.47(a)(1)(iv) to identify:

"Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG-0933 current on the date six months prior to application and which are technically relevant to the design."

NUREG-0933, "A Prioritization of Generic Safety Issues," through Supplement 25 identifies hundreds of issues. The issues tabulated in Supplement 25 were reviewed to determine which issues are technically relevant to the AP1000 design. In this review process, the following screening criteria were applied:

a. Issue has been prioritized as **Low**, **Drop**, or has not been prioritized.

b. Issue is not an AP1000 design issue. Issue is applicable to GE, B&W, or CE designs only.

c. Issue resolved with no new requirements.

d. Issue is not a design issue (Environmental Issue, Licensing Issue, Regulatory Impact Issue, or covered in an existing NRC program).

e. Issue superseded by one or more issues.

f. Issue is not an AP1000 design certification issue. Issue is applicable to NTOL plants only, responsibility of combined license applicant, or issue is limited to current generation operating plants.

Issues meeting one or more of the preceding screening criteria were screened out of the review process as issues that are not applicable to the AP1000 design. The remaining issues fall into one of the following two categories:

g. Issue is resolved by establishment of new regulatory requirements and/or guidance.

h. Issue is unresolved pending generic resolution (e.g., prioritized as **High**, **Medium**, or possible resolution identified).

Table 1.9-2 identifies the results of the screening review. For those issues identified as relevant to the AP1000 design (i.e., issues screened as **g** or **h**), Table 1.9-2 identifies the DCD subsection that addresses the issue.

1.9.4.2 AP1000 Resolution of Unresolved Safety Issues and Generic Safety Issues

1.9.4.2.1 TMI Action Plan Issues

TMI Action Plan issues that were not incorporated in 10CFR50.34(f) are addressed in the following. Those issues incorporated into 10CFR50.34(f) are addressed in subsection 1.9.3.

I.D.5(2) Plant Status and Post-Accident Monitoring Discussion:

TMI action plant item I.D.5(2) addresses the need to improve the operators' ability to prevent, diagnose and properly respond to accidents. The emphasis is on the information needs (i.e., indication of plant status) of the operator. This issue was resolved with the issuance of Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident."

AP1000 Response:

The AP1000 conforms to and meets the intent of Regulatory Guide 1.97. Regulatory Guide 1.97 provides the requirements for post-accident monitoring of nuclear reactor safety parameters, including plant process parameters important to safety and the monitoring of effluent paths and plant environs for radioactivity. These guidelines include definition and categorization of plant variables that are available to the main control room operators for monitoring the plant safety status following a design basis event.

For the AP1000, an analysis is conducted to identify the appropriate variables and to establish the appropriate design basis and qualification criteria for instrumentation used by the operator for monitoring conditions in the reactor coolant system, the secondary heat removal system, the containment, and the systems used for attaining a safe shutdown condition, as discussed in Section 7.5.

The instrumentation is used by the operator to monitor and maintain the safety of the plant during operating conditions, including anticipated operational occurrences and accident and post-accident conditions. A set of plant parameters identified according to the Regulatory Guide 1.97 guidelines are processed and displayed by the qualified data processing system (QDPS), which is discussed in subsection 18.8. The verification and validation (V&V) of the QDPS complies with the V&V process described in Section 18.11.

I.D.5(3) On-Line Reactor Surveillance System

Discussion:

TMI action plan item I.D.5(3) addresses the benefit to plant safety and operations of continuous on-line automated surveillance systems. Continuous on-line surveillance systems that automatically monitor reactors can assist plant operations by providing diagnostic information which can predict anomalous behavior.

Various methods of on-line reactor surveillance have been used, including neutron noise monitoring in boiling water reactors (BWRs) to detect internals vibration, and pressure noise surveillance at TMI-2 to monitor primary loop degasification.

AP1000 Response:

The AP1000 reactor coolant pressure boundary is monitored for leaks from the reactor coolant and associated systems by a variety of components located in multiple systems. The leak detection system provides information permitting the plant operators to take corrective action if any detected leakage exceeds technical specifications. The leak detection system is designed according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 30. The system provides a means to detect and, to the extent practical, to identify the source of the reactor coolant pressure boundary leakage. DCD subsection 5.2.5 provides further discussion of leak detection.

A digital metal impact monitoring system (DMIMS) monitors the reactor coolant system for the presence of loose metallic parts. This system conforms with the guidance provided in Regulatory Guide 1.133, Rev. 1, May 1981. An advanced microprocessor-based system, employing digital technology, automatically actuates audible and visual alarms if a signal exceeds the preset alarm level.

I.F.1 Expand Quality Assurance List

Discussion:

Item I.F.1 addressed the issue of systems that are "important to safety" that are not on the Quality Assurance List. The suggestion was made that equipment important to safety be ranked and that ranking used to determine systems that should be added to the Quality Assurance List. This approach has not been implemented by the NRC on either a generic or cases-by case basis. In NUREG-0933 this item was classified as resolved with no additional requirements established.

AP1000 Response:

The requirements of 10 CFR Appendix B apply to safety-related systems and components. See subsection 3.2.2 for a discussion of the AP1000 equipment classification system and the associated quality assurance requirements, including requirements for nonsafety-related systems.

I.G.1 Training Requirements

Discussion:

Item I.G.1 included the issue of natural circulation testing for use as input into operator training.

AP1000 Response:

For the AP1000, natural circulation heat removal using the steam generators is not safety-related, as in current plants. This safety-related function is performed by the passive residual heat removal system. Natural circulation heat removal via the passive residual heat removal heat exchanger is tested for every plant during hot functional testing. This testing of passive residual heat removal system meets the intent of the requirement to perform natural circulation testing and the results of this testing is factored into the operator training.

For the AP1000, the tests outlined below are contained in the AP1000 initial test plan and demonstrate the effectiveness of natural circulation cooling.

1. During hot functional testing, prior to fuel load, with the reactor coolant pumps not running and no onsite power available, the heat removal capability of the passive residual heat removal heat exchanger with natural circulation flow is verified (See subsection 14.2.9.1.3, item e).

2. After fuel loading, but prior to criticality, with the reactor system at no-load operating temperature and pressure and all reactor coolant pumps operating, the depressurization rate is determined by de-energizing the heaters and pressure is further reduced through use of sprays (See subsection 14.2.10.1.19).

3. After criticality is achieved and the plant is at ~ 3% power, the plant is placed in a natural circulation mode by tripping all reactor coolant pumps and observing the plant response using the steam generators (See subsection 14.2.10.3.6) and then using the PRHR (see subsection 14.2.10.3.7) as the primary heat sinks. These tests are performed for the first plant only.

4. A loss-of-offsite power test is performed with the plant at minimum power level supplying normal house loads. The turbine is tripped and the plant is placed in a stable condition using batteries and the diesel generator (See subsection 14.2.10.4.26).

5. Data obtained from the first plant only natural circulation tests using the steam generators and PRHR is provided for operator training on a plant simulator at the earliest opportunity. Operating training for subsequent plants is also obtained while performing the hot functional PRHR natural circulation test described in item 1 above.

This response as modified for the AP1000 design is consistent with the response to NUREG-0737, action item I.G.1 which provided a proposal for low power testing of existing and future Westinghouse pressurized water reactors in Attachment 4 to letter NS-EPR-2465 from Westinghouse (E. P. Rahe) to the NRC (H. R. Denton) dated July 8, 1981.

I.G.2 Scope of Test Program

Discussion:

TMI Action Plan Items I.G.2 recommended additional testing during preoperational and startup programs to search for anomalies in a plants response to transients. The Standard Review Plan, Section 14 was revised to provide additional guidance for preoperational and startup test programs.

AP1000 Response:

The program plan for preoperational and startup testing of the AP1000 is in Section 14.2. This section addresses the Standard Review Plan, Section 14. The conformance with Standard Review Plan, Section 14 is outlined in AP1000 Compliance with SRP Acceptance Criteria, WCAP-15799.

II.E.1.3 Update Standard Review Plan and Develop Regulatory Guide

Discussion:

This item was a requirement to update Section 10.4.9 of the Standard Review Plan to address the requirements of Items II.E.1.1 and II.F.1.2 for auxiliary feedwater systems. Standard Review Plan 10.4.9 was revised and this issue is classified as resolved.

AP1000 Response:

The AP1000 does not have a safety-related auxiliary feedwater system. For conformance of the AP1000 with Items II.E.1.1 and II.E.1.2 see the write-up for (1)(ii) and (2)(xii) in subsection 1.9.3. For conformance with Standard Review Plan Section 10.4.9 see WCAP-15799.

II.E.6.1 Test Adequacy Study

Discussion:

This item was intended to establish the adequacy of requirements for safety-related valve testing. Subsequent to this item, expanded requirements were written into the ASME OM Code for valve testing.

AP1000 Response:

The AP1000 is designed for an in-service test program in accordance with the ASME OM Code. See subsection 3.9.6 for additional information on the in-service testing program plan.

II.K.1(10) Review and Modify Procedures for Removing Safety-related Systems from Service

Discussion:

This item required operating plants to review and modify (as required) their procedures for removing safety-related systems from service to assure operability status is known.

AP1000 Response:

DCD Section 13.5 describes the AP1000 procedure development, preparation, and responsibility.

II.K.1(13) Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items.

Discussion:

This item required that operating plants propose technical specification changes to address Bulletin items.

AP1000 Response:

The AP1000 Technical Specifications (Section 16.1) are based on and were reviewed against the Westinghouse Standard Technical Specifications, which incorporated the requirements of the bulletins for the TMI Action Plan.

II.K.1(17) Trip PZR Level Bistable So That Low Pressure Will Initiate Safety Injection

Discussion:

This item required operating licensees and operating license applicants with Westinghouse designed nuclear steam supply systems to trip the pressurizer level bistable so that the pressurizer low pressure (rather than the pressurizer low pressure and pressurizer low level coincidence) would initiate safety injection.

AP1000 Response:

This issue does not apply to AP1000. The AP1000 does not rely on coincident low pressurizer pressure and low pressurizer level for actuation. See Section 6.3 for a discussion of actuation of the passive core cooling system.

II.K.1(24) Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and Reactor Coolant Pump Trip

Discussion:

This item requires analyses to provide the basis for the comparison of analytical methods.

AP1000 Response:

The analyses documented in Chapter 15 cover a range of small break sizes. The AP1000 automatically trips the reactor coolant pump on an SI signal. The need to look at time lapses between reactor trip and pump trip is not required.

II.K.3(5) Automatic Trip of Reactor Coolant Pumps

Discussion:

This item requires that operating plants and operating plant applicants study the need for automatic trip of reactor coolant pumps and to modify procedures of designs as appropriate.

AP1000 Response:

The AP1000 design provides for an automatic trip of the reactor coolant pumps on actuation of the passive core cooling system. This trip is provided to prevent reactor coolant pump interaction with the operation of the core makeup tank. See Section 6.3 for additional information.

II.K.3(9) Proportional Integral Derivative Controller Modification

Discussion:

TMI action plan item II.K.3(9) required all Westinghouse plants to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not include power-operated relief valves. See subsections 5.1.2 and 5.2.2 for additional information.

1.9.4.2.2 Task Action Plan Items

A-1 Water Hammer

Discussion:

Generic Safety Issue A-1 was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and piping, emergency core cooling systems, residual heat removal systems, containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage has been relatively minor and involved pipe hangers and restraints. However, several incidents have resulted in piping and valve damage. This item was originally identified in NUREG-0371, (Reference 4) and was later determined to be an Unresolved Safety Issue.

AP1000 Response:

Specific sections of the Standard Review Plan (NUREG-0800) address criteria for mitigation of water hammer concerns. The applicable Standard Review Plan sections as well as information provided in NUREG-0927 (Reference 5) were reviewed. The AP1000 meets the water hammer provisions as specified. The discussion that follows provides a brief description of selected systems identified as being subject to water hammer occurrences and special design features that mitigate or prevent water hammer damage.

Design features are incorporated as appropriate to prevent water hammer damage in applicable systems including steam generator feedrings and piping, passive core cooling system, passive residual heat removal system, service water system, feedwater system, and steam lines.

Water hammer issues are considered in the design of the AP1000 passive core cooling system. The passive core cooling system design includes a number of design features specifically to prevent or mitigate water hammer.

The automatic depressurization system operation uses multiple, sequenced valve stages to provide a relatively slow, controlled depressurization of the reactor coolant system, which helps to reduce the potential for water hammer.

Once the depressurization is complete, gravity injection from the in-containment refueling water storage tank is initiated by opening squib valves and then check valves, which reposition slowly. Gravity injection flow actuates slowly, without water hammer, as the pressure differential across the gravity injection check valves equalizes, and the valves open and initiate flow.

The passive residual heat removal heat exchanger is normally aligned with an open inlet valve and closed discharge valves. This alignment keeps the system piping at reactor coolant system pressure, preventing water hammer upon initiation of flow through the heat exchanger. Instrumentation is provided at the system high point to detect a void in the system.

The core makeup tanks are normally aligned with an open inlet line from the reactor coolant cold leg to keep the tanks at reactor coolant system pressure. This alignment keeps the system piping at reactor coolant pressure, preventing water hammer upon initiation of flow through the tank. In addition, instrumentation is provided at each high point to detect voids within the system. Section 6.3 of the DCD provides additional information on the passive core cooling system.

The potential for water hammer in the feedwater line is minimized by the improved design and operation of the feedwater delivery system. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundles and below the normal water level by a top discharge spray tube feedring. The feedring is welded to the feedwater nozzle to limit the potential for inadvertent draining. The layout of the feedwater line is consistent with industry standard recommendations to reduce the potential of a steam generator water hammer.

The startup feedwater system is a nonsafety-related system that provides feedwater during normal plant startup, shutdown, and hot standby. The startup feedwater line is separate from the main feedwater line and therefore does not contribute to the potential of water hammer in the feedwater piping or steam generator feedring.

The main steam line drains are designed to remove accumulated condensate from the main steam lines and to maintain the turbine bypass header at operating temperature during plant operation. The system is designed to accommodate drain flows during startup, shutdown, transient, and normal operation to protect the turbine and the turbine bypass valves from water slug damage.

A-2 Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

Discussion:

Generic Safety Issue A-2 pertains to asymmetric loadings that could act on a pressurized water reactor's primary system as the result of a postulated double-ended rupture of the piping in the primary coolant system. The magnitude of these loads is potentially large enough to damage the supports of the reactor vessel, the reactor internals, and other primary components of the system. Therefore, the NRC initiated a generic study to develop criteria for an evaluation of the response of the primary systems in pressurized water reactors to these loads.

AP1000 Response:

The use of mechanistic pipe break criteria permits elimination of the evaluation of dynamic effects of sudden circumferential and longitudinal pipe breaks in the structural analysis of structures, systems, and components. General Design Criterion 4 allows the use of analyses to eliminate from the design basis the dynamic effects of pipe ruptures postulated at locations defined in subsection 3.6.2. Dynamic effects include jet impingement, pipe whip, jet reaction forces on other portions of the piping and components, subcompartment pressurization including reactor cavity asymmetric pressurization transients, and traveling pressure waves from the depressurization of the system.

The AP1000 reactor coolant loop and pressurizer surge line are designed in accordance with mechanistic pipe break criteria. In addition, other high energy ASME Code, Section III, Class 1 and 2 piping of 6 inches and greater nominal diameter is evaluated against leak-before-break criteria. The evaluation methodology is described in subsection 3.6.3 and Appendix 3B.

A-3 Steam Generator Tube Integrity

Discussion:

Pressurized water reactor steam generator tube integrity is subject to various degradation mechanisms, including corrosion-induced wastage, cracking, reduction in tube diameter, denting, (which leads to primary side stress corrosion cracking), vibration-induced fatigue cracks, and wear or fretting due to loose parts in the secondary system. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Steam generator tube integrity concerns for the three steam generator suppliers, Westinghouse, Combustion Engineering, and Babcock and Wilcox, are addressed by an integrated NRC program for Generic Safety Issues A3, A4, and A5. This program addresses the areas of steam generator integrity, plant systems response, human factors, radiological consequences, and the response of various organizations to a steam generator tube rupture.

AP1000 Response:

The AP1000 steam generators are designed in accordance with the recommendations of Generic Letter 85-02 and NUREG-0844 (References 6 and 7). The AP1000 steam generator is equipped with a number of features to enhance steam generator tube performance and reliability. These features are described in subsection 5.4.2.

A-9 Anticipated Transients Without Scram

Discussion:

Generic Safety Issue A-9 was resolved with the publication of 10 CFR 50.62. This regulation sets forth the requirements for reduction of risks from anticipated transients without scram.

AP1000 Response:

The AP1000 complies with the requirements of 10 CFR 50.62 except that the AP1000 does not have a safety-related auxiliary feedwater system. In lieu of the automatic initiation of the auxiliary feedwater system under conditions indicative of an ATWS as required by 10 CFR 50.62 (c)(1), the AP1000 automatically initiates the passive residual heat removal system as discussed in Section 6.3.

A discussion of the AP1000 design features used to address the probability of an ATWS is presented in subsection 1.9.5 and Section 7.7.

A-11 Reactor Vessel Materials Toughness

Discussion:

Generic Issue A-11 addresses a concern with the reduction of reactor vessel fracture toughness as plants accumulate more and more service time. 10 CFR 50, Appendix G provides requirements for reactor vessel material toughness.

AP1000 Response:

The AP1000 reactor vessel design complies with the requirements of 10 CFR 50, Appendix G and includes numerous features to reduce neutron fluence, enhance material toughness at low temperature and eliminate weld seams in critical areas. Material requirements are provided in subsection 5.3.2. Pressure and temperature limits are provided in subsection 5.3.3.

A-12 Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

Discussion:

Generic Safety Issue A-12 addresses a concern with the potential for lamellar tearing of steam generator and RCP support material. NUREG-0577 (Reference 8) categorizes operating plants relative to the adequacy of the plant's steam generator and reactor coolant pump supports with respect to fracture toughness.

AP1000 Response:

The steam generator and reactor coolant pump supports are described in subsection 5.4.10. The supports are designed in accordance with subsection NF of Section III of the ASME Code. Design and fabrication of these supports in accordance with Subsection NF requirements provide acceptable fracture toughness of materials, and conform with NUREG-0577.

A-13 Snubber Operability Assurance

Discussion:

Generic Issue A-13 addresses snubber operability concerns. Snubbers are utilized primarily as seismic and pipe whip restraints at nuclear power plants. Their safety function is to operate as rigid supports for restraining the motion of attached systems or components under rapidly applied load conditions such as earthquakes, pipe breaks, and severe hydraulic transients.

Operating experience reports show that a substantial number of snubbers have leaked hydraulic fluid and that the rejection rate from functional testing and inspection is high. This has led to an NRC and ACRS concern regarding the effect of snubber malfunctions on plant safety.

AP1000 Response:

The use of snubbers is minimized in the AP1000. Gapped support devices, leak-before-break considerations, and state-of-the-art piping analysis methods are used to minimize the use of snubbers. Snubbers applied in safety-related applications are constructed to ASME Code, Section III, Subsection NF as discussed in DCD subsection 3.9.3.4.3.

A-17 Systems Interactions in Nuclear Power Plants

Discussion:

This item addresses the potential systems interactions among systems including safety-related and nonsafety-related structures, systems, and components. There can be unintended and unrecognized dependencies among structures, systems, and components. A number of specific types of interactions have been addressed in other generic safety issues and NRC staff activities. These include guidance for inclusion of internal flooding in the IPE program, requirements that address seismically-induced systems interactions, and evaluation of electric power supplies for electric power reliability. NUREG-0933 classifies this item as resolved with no new requirements.

AP1000 Response:

In addition to addressing the specific system interaction guidance mentioned above, the AP1000 was the subject of a systematic evaluation of potential adverse systems interactions documented in WCAP-15992, "AP1000 Adverse Systems Interactions Evaluation Report" (Reference 69).

A-24 Qualification of Class 1E Safety-Related Equipment

Discussion:

Generic Issue A-24 was resolved with the publication of 10 CFR 50.49, prescribing aging and testing for synergistic effects. The NRC has also issued Revision 1 to Regulatory Guide 1.89 for comment. The proposed revision describes a method acceptable to the NRC staff to demonstrate compliance with the requirements of 10 CFR 50.49.

AP1000 Response:

The AP1000 environmental qualification methodology described in Appendix 3D is based on the generic Westinghouse qualification program approved by the NRC. The Westinghouse methodology addresses the requirements of General Design Criteria 4 and 10 CFR 50.49, as well as the guidance of Regulatory Guide 1.89 and IEEE Standard 323-1974. See Appendix 1A and Reference 9.

A-25 Non-Safety Loads on Class 1E Power Sources

Discussion:

Generic Issue A-25 addresses whether nonsafety-related loads should be allowed to share Class 1E power sources with safety-related plant systems. Past regulatory practice has allowed the connection of nonsafety-related loads in addition to the required safety loads to Class 1E power sources by imposing some restrictions. The purpose of this issue is for the NRC to determine whether the reliability of the Class 1E power sources is significantly affected by the sharing of safety and nonsafety-related loads.

The NRC considers this issue as technically resolved with the issuance of Revision 2 to Regulatory Guide 1.75. This regulatory guide includes special requirements for connection of nonsafety-related loads to a Class 1E source.

AP1000 Response:

The AP1000 conforms with the criteria of Regulatory Guide 1.75 with minor exceptions (see Appendix 1A and IEEE 384-1974). The AP1000 safety-related power source is the Class 1E dc and UPS system, which supplies power to the ac inverters for the plant instrumentation and control systems. The system also provides power to dc loads associated with the four protection channels and the accident monitoring system. Non-Class 1E loads powered from Class 1E sources are limited to loads that need connection to a reliable power source. No Credible failure of non-Class 1E equipment or systems will degrade the Class 1E system below an acceptable level. Subsection 8.3.2.1.1 provides a discussion on the Class 1E power source.

A-26 Reactor Vessel Pressure Transient Protection

Discussion:

Generic Issue A-26 addresses the need to provide reactor vessel overpressure protection whenever plants are in a cold shutdown condition. Branch Technical Position RSB 5-2 establishes the current NRC criteria for a low-temperature overpressurization protection system.

AP1000 Response:

The AP1000 conforms with the criteria established in Branch Technical Position RSB 5-2. The AP1000 pressurizer is sized to accommodate most pressure transients. Overpressure protection for the reactor coolant system is provided by either the pressurizer safety valves or the normal residual heat removal relief valves, as described in subsection 5.2.2.

A-28 Increase in Spent Fuel Pool Storage Capacity

Discussion:

Generic Issue A-28 addresses the safety significance of damage to spent fuel, primarily from a lack of adequate cooling, that could result in the release of radioactivity.

AP1000 Response:

The AP1000 incorporates the NRC criteria. The heat load is evaluated for the spent fuel storage capacity.

A-29 Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Description

This item addresses potential methods to reduce vulnerability to sabotage. The NRC staff concluded that existing requirements dealing with plant physical security, controlled access to vital areas, screening for reliable personnel appear to be effective. This item was resolved with no new requirements.

AP1000 Response:

The passive systems in the AP1000 provided to mitigate the effects of potential accidents may have an inherent advantage when considering potential acts of sabotage compared to the active systems in operating plants. The AP1000 includes provisions for access control to the vital area. The provisions for security are discussed in the AP1000 Security Design Report and outlined in Section 13.6.

A-31 Residual Heat Removal Requirements

Discussion:

Generic Issue A-31 addresses the desire for plants to be able to go from hot-standby to cold-shutdown conditions (when this is determined to be the safest course of action) under an accident condition. The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident has been typically interpreted as achieving a hot standby condition (the reactor is shut down, but system temperature and pressure are at or near normal operating values). There are events that require eventual cooldown and long-term cooling to perform inspection and repairs.

AP1000 Response:

The AP1000 employs safety-related core decay heat removal systems that establish and maintain the plant in a safe shutdown condition following design basis events. It is not necessary that these passive systems achieve cold shutdown as defined by Regulatory Guide 1.139.

The AP1000 complies with General Design Criteria 34 by using a more reliable and simplified system design. The passive core cooling system is employed for both hot-standby and long-term cooling modes. Hot-standby conditions are achieved immediately and a temperature of 420°F is reached within 36 hours. Reactor pressure is controlled and can be reduced to about 250 psig. The passive residual heat removal system provides a closed cooling system to maintain long-term core cooling. Passive feed and bleed cooling, using the passive injection features for the feed and the automatic depressurization system for bleed, provides another closed-loop safety-related cooling capability. This capability eliminates dependency on open-loop cooling systems, which have limited ability to remain in hot standby for long-term core cooling. See Section 7.4 for a discussion of safe shutdown and Section 6.3 for a description of the passive core cooling system.

Since the passive core cooling system maintains safe conditions indefinitely, cold shutdown is necessary only to gain access to the reactor coolant system for inspection or repair. On the AP1000, cold shutdown is accomplished by using non-safety-related systems. These systems are highly reliable. They have similar redundancy as current generation safety-related systems and are supplied with ac power from either onsite or offsite sources. See subsection 5.4.7 for a description of the normal residual heat removal system and subsection 7.4.1.3 for a discussion of cold shutdown achieved by use of non-safety-related systems.

A-35 Adequacy of Offsite Power Systems

Discussion:

Generic Issue A-35 addresses the susceptibility of safety-related electric equipment to offsite power source degradation. The NRC considers this issue as technically resolved with the issuance of the Standard Review Plan, Section 8.3.1 criteria specified in Appendix A, Branch Technical Position BTP PSB 1, "Adequacy of Station Electric Distribution System Voltages."

AP1000 Response:

The AP1000 ac power system is discussed in subsections 8.1 through 8.3. The AP1000 does not require any ac power source to achieve and maintain safe shutdown.

A-36 Control of Heavy Loads Near Spent Fuel

Discussion:

Generic Issue A-36 addresses the need to review requirements, facility designs, and Technical Specifications regarding the movement of heavy loads near spent fuel. The NRC has documented its technical position on this issue in NUREG-0612 (Reference 10) and that issued Standard Review Plan, Section 9.1.5, which includes NUREG-0612 as a part of the review plan.

AP1000 Response:

The AP1000 design conforms to NUREG-0612 and Standard Review Plan, Section 9.1.5. Light load handling systems are described in subsection 9.1.4, and overhead heavy-load handling systems are described in subsection 9.1.5.

A-39 Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments

Discussion:

Generic Issue A-39 addresses operation of BWR primary system pressure relief valves whose operation can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperatures. This USI was resolved with the issuance of SRP Section 6.2.1.1.C and a series of NUREG reports.

Generic Issue A-39 is not directly applicable to the AP1000. However, the AP1000 in-containment refueling water storage tank (IRWST) has some functional similarity to a suppression pool when the automatic depressurization system (ADS) is actuated.

AP1000 Response:

The AP1000 in-containment refueling water storage tank design includes consideration of loads due to automatic depressurization system operation. The effect of hydrodynamic loads is addressed in DCD subsection 3.8.3.4.2.

A-40 Seismic Design Criteria - Short Term Program

Discussion:

Generic Issue A-40 addresses a desire to identify and quantify conservatism in the seismic design process. The Standard Review Plan, Section 3.7 provides clarification of development of site-specific spectra, justification for use of single synthetic time-history by power spectral density function, location and reductions of input ground motion for soil-structure interaction, and design of above-ground vertical tanks. The revised provisions are used for margin studies and re-evaluations or individual plant examination for external events.

AP1000 Response:

The AP1000 conforms to the criteria outlined in the Standard Review Plan, Section 3.7. The seismic design criteria and seismic evaluation methodology are described in Section 3.7.

The AP1000 employs generic, enveloping seismic design criteria and applies established seismic evaluation methodology that complies with current regulations and regulatory guidance. For sites having specific characteristics outside the range of the selected parameters, the AP1000 is evaluated to demonstrate acceptability to the site-specific characteristics.

A-43 Containment Emergency Sump Performance

Discussion:

Generic Issue A-43 addresses technical concerns as follows:

- Pressurized water reactor sump (or boiling water reactor residual heat removal system suction intake) hydraulic performance under post-loss-of-coolant accident adverse conditions resulting from potential vortex formation, air ingestion, and subsequent pump failure

- The possible transport of large quantities of insulation debris generated by a loss-of-coolant accident resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling

- The capability of residual heat removal and containment spray system pumps to continue pumping when subjected to possible air, debris, or other effects, such as particulate ingestion on pump seal and bearing systems

AP1000 Response:

Air ingestion, vortexing, and debris blockage are not significant concerns for the AP1000. Containment recirculation includes sump screens that conform to the criteria specified in Regulatory Guide 1.82. The recirculation screens have a large cross-sectional area to reduce the fluid flow velocity through the screen and to provide a large screening area to accommodate accumulated debris. Horizontal plates located above the recirculation screens preclude debris being deposited in the water directly adjacent to the screens. Pipe subject of loss of coolant pipe breaks and in the vicinity of these breaks use reflective metallic insulation to preclude the generation of fibrous insulation debris. See subsection 6.3.2.2.7 for additional information on the design of the screens and limits on use of fibrous insulation.

Since the AP1000 design does not use pumps to provide safety injection flow, the passive core cooling system injection flow rates are substantially lower than those for plants with pumped injection flow. This results in lower fluid flow velocities through the screens, reducing the potential to draw debris into the sump screens.

The containment recirculation sump piping inlet is located slightly above the compartment floor, which is substantially below the expected flood-up water level. This precludes air ingestion in the piping since recirculation does not initiate until the flood-up water level is well above the piping inlet.

The elimination of pumps also eliminates concerns about the effects on safety injection capability for vortexing, air ingestion, and blockage effects on pump net positive suction head.

The AP1000 includes the capability to use nonsafety-related normal residual heat removal pumps to take a suction from the containment recirculation sump to provide reactor coolant system injection. The sump screen design addresses concerns with screen debris, vortexing, and air ingestion.

Section 6.3 provides additional information on the operation of the passive core cooling system. Appendix 1A describes conformance with Regulatory Guide 1.82. Section 6.2 provides additional information on the containment recirculation sump.

A-44 Station Blackout

Discussion:

Generic Issue A-44 was resolved with the publication of 10 CFR 50.63, which provides requirements that light-water-cooled nuclear power plants be able to withstand for a specified duration and recover from a station blackout. It specifies that an alternate ac power source constitutes acceptable capability to withstand station blackout provided an analysis is performed that demonstrates that the plant has this capability from the onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate.

10 CFR 50.2 for the alternate ac source notes that the alternate ac power source must have sufficient capability and reliability for operation of all systems required for coping with station blackout for the time required to place and maintain the plant in safe shutdown.

AP1000 Response:

AC electrical power is not needed to establish or maintain a plant safe shutdown condition for the AP1000. The ac power system is discussed in Chapter 8. In addition, two nonsafety-related standby diesel generators are provided as alternate sources of electrical power to nonsafety-related active systems that provide a defense-in-depth function.

A-46 Seismic Qualification of Equipment in Operating Plants

Discussion:

Generic Issue A-46 addresses the variability among operating plants in the margins of safety provided in equipment to resist seismically induced loads and perform the intended safety functions. The NRC believes that the seismic qualification of equipment in operating plants must, therefore, be reassessed to confirm the ability to bring the plant to a safe shutdown condition when it is subject to a seismic event.

AP1000 Response:

This issue applies to operating plants and, as such, does not specifically apply to the AP1000, which is designed in accordance with current seismic requirements. The seismic Category I mechanical and electrical equipment utilized for the AP1000 is qualified in accordance with the AP1000 qualification methodology discussed in Section 3.10. The methodology is based on the generic Westinghouse qualification program previously approved by the NRC. This methodology addresses IEEE Standard 344-1987 (Reference 13) and Regulatory Guide 1.100. See subsection 1.9.1 (Appendix 1A).

A-47 Safety Implications of Control Systems

Discussion:

Generic Issue A-47 addresses the safety impact of non-safety-related control systems on plant dynamics. Instrumentation and control systems used by nuclear plants comprise safety-related protection systems and nonsafety-related control systems. Safety-related systems are used to trip the reactor when specified parameters exceed allowable limits and to protect the core from overheating by initiating emergency core cooling systems. Nonsafety-related control systems are used to maintain the plant within prescribed parameters during shutdown, startup, normal load, and varying power operation. Nonsafety-related systems are not relied on to perform any safety functions during or following postulated accidents, but are used to control plant processes.

AP1000 Response:

For the AP1000, control system failures are considered as potential initiating events. The analyses of these transients demonstrate that the consequences of such failures are bounded by ANS Condition II criteria. No design basis failure of a control system violates Condition II criteria.

The integrated control system for the AP1000 obtains certain control input signals from signals used in the integrated protection system. With the integrated control and protection system, functional independence of the control and protection systems is maintained by providing a signal selection device in the control system for those signals used in the protection system. The purpose of the signal selection device is to prevent a failed signal, caused by the failure of a protection channel, from resulting in a control action that could lead to a plant condition requiring that protective action. The signal selection device provides this capability by comparing the redundant signals and automatically eliminating an aberrant signal from use in the control system. This capability exists for bypassed sensors or for sensors whose signals diverge from the expected error tolerance.

The plant control system incorporates design features such as redundancy, automatic testing, and self-diagnostics to prevent challenges to the protection and safety monitoring system. Chapter 7 provides a discussion of the AP1000 instrumentation and controls. The surveillance requirements for the main and startup feedwater control are found in Technical Specifications 3.7.3 and 3.7.7.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Discussion:

Generic Issue A-48 addresses postulated light water reactor accidents resulting in a degraded or melted core that could result in the generation and release to the containment of large quantities of hydrogen. One source of hydrogen is from the reaction of the zirconium fuel cladding with the steam at high temperatures. The NRC requires design provisions for handling hydrogen releases associated with rapid reaction of a large portion of fuel cladding (10 CFR 50.44 and 10 CFR 50.34).

AP1000 Response:

The AP1000 design complies with the provisions of draft changes to 10 CFR 50.44 and 10 CFR 50.34 (f). The mechanisms used to monitor and control hydrogen inside containment are discussed in subsection 6.2.4.

A-49 Pressurized Thermal Shock

Discussion:

Generic Issue A-49 addresses transients and accidents postulated to occur in pressurized water reactors that can result in severe overcooling (thermal shock) of the reactor vessel, concurrent with high pressure. In these pressurized thermal shock events, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall that produces a thermal stress with maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress varies with the rate of change of temperature and is compounded by coincident pressure stresses.

As long as the fracture resistance of the reactor vessel material is relatively high, these events are not expected to cause vessel failure. The fracture resistance of the reactor vessel material decreases with the integrated exposure to fast neutrons. The rate of decrease is dependent on the chemical composition of the vessel wall and weld materials.

AP1000 Response:

The AP1000 complies with the requirements of 10 CFR 50.61. Material requirements and pressure-temperature limits are discussed in subsections 5.3.2 and 5.3.3.

B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Discussion:

Part I - Ductility of Two-Way Slabs and Shells

Generic Issue B-5 involved a concern over the lack of information on the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension, flexure, and shear. The

NRC Staff concluded that there is sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis.

Part II - Buckling Behavior of Steel Containments

Generic Issue B-5 involves a concern over the lack of a uniform, well defined approach for design evaluation of steel containments. Of particular interest was potential instability of the shell during dynamic loadings. Based on the conclusion of the NRC Staff that existing steel containments had adequate margins against buckling and that the issue of steel containment buckling had very little safety impact, this item was classified as resolved with no new requirements.

AP1000 Response:

The design requirements and analysis methods used for two-way reinforced concrete slabs and for the steel containment are outlined in DCD Section 3.8.

B-17 Criteria for Safety-Related Operator Actions

Discussion:

Generic Issue B-17 addresses the development of a time criterion for safety-related operator actions including a determination of whether or not automatic actuation is required. The evaluation of this issue includes Issue 27, Manual versus Automated Actions.

AP1000 Response:

The AP1000 automatically initiates the safety-related actions required to protect the plant during design basis events. The plant systems are designed to provide the required information to the operator to monitor plant conditions and to evaluate the performance of the safety-related passive systems, as well as the nonsafety-related active systems. The active systems are designed to automatically actuate and provide defense-in-depth for various plant events, to preclude unnecessary actuation of the safety-related passive systems. The plant design also provides the capability for a backup manual initiation of both the safety-related systems and the nonsafety-related defense-in-depth systems.

As described in Chapter 15, the AP1000 safety systems maintain the plant in a safe condition following design basis events. For the design basis events described in Chapter 15, this is accomplished without the need for operator action for up to 72 hours. Operator action is planned and expected during plant events to achieve the most effective plant response consistent with event conditions and equipment availability. For events where operator action is taken, the plant design maximizes the time available to complete actions for events. For example, during a steam generator tube rupture, no operator action is required to establish safe shutdown conditions or prevent steam generator overfill. It is expected that the main control room operators take actions similar to those taken in current plants to identify and isolate the faulted steam generator and to stabilize plant conditions.

For events where operator actions are taken, the AP1000 design is based on previous experience and the guidance of ANSI 58.8-1984 (Reference 21). At least 30 minutes is available following design basis events for the operator to initiate planned actions.

B-22 LWR Fuel

Discussion:

Generic Issue B-22 addresses the reliability of fuel behavior predictions during normal operation and postulated accidents. Standard Review Plan, Section 4.2 provides detailed NRC criteria for the design of fuel and core components.

AP1000 Response:

The AP1000 reactor core design complies with the Standard Review Plan, Section 4.2. See Section 4.2 for a discussion of the fuel system design.

B-29 Effectiveness of Ultimate Heat Sinks

Discussion:

Generic Issue B-29 addresses NRC confirmation of currently used mathematical models for prediction of ultimate heat sink performance by comparing model performance with field data and development of better guidance regarding the criteria for weather record selection to define ultimate heat sink design basis meteorology.

The NRC considers this issue to be technically resolved with the publication of three reports: NUREG-0693, NUREG-0733, and NUREG-0858 (References 23, 24 and 25).

AP1000 Response:

The AP1000 passive containment cooling system complies with Standard Review Plan, Section 9.2.5 by providing passive decay heat removal that transfers heat to the atmosphere, which is the ultimate heat sink for accident conditions. The passive containment cooling system is described in subsection 6.2.2.

B-32 Ice Effects on Safety-Related Water Supplies

Discussion:

Generic Issue B-32 addresses the potential effects of extreme cold weather and ice buildup on the reliability of various plant water supplies. Current NRC criteria are provided in Standard Review Plan, Section 2.4.7, "Ice Effects."

AP1000 Response:

Subsection 6.2.2 describes the ultimate heat sink design and discusses the features that prevent freezing in the passive containment cooling system.

B-36 Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems

Discussion:

Generic Issue B-36 addresses the development of revisions to current guidance and technical positions regarding engineered safety features and normal ventilation system air filtration and adsorption units. The NRC considers this issue technically resolved with the issuance of Revision 2 to Regulatory Guide 1.52 and Revision 1 to Regulatory Guide 1.140.

AP1000 Response:

The AP1000 main control room emergency habitability system (VES) includes a passive filtration system that is contained entirely within the main control room envelope. Regulatory Guide 1.52 was written for active safety-related filtration systems. To the extent applicable, system design criteria are established in accordance with Regulatory Guide 1.52 Revision 3. The passive filtration portion of the AP1000 VES contains no active equipment.

B-53 Load Break Switch

Discussion:

Generic Issue B-53 addresses the use of the generator load break switch for isolating the generator from the step-up transformer following turbine trip. Plant designs that utilize generator load circuit breakers to satisfy the requirement for an immediate access circuit stated in General Design Criterion 17, "Electric Power Systems," must prototype-test the generator load circuit breaker to demonstrate functional capability.

AP1000 Response:

The AP1000 design incorporates a generator load circuit breaker to provide a reliable source of ac power to the electrical systems. Exceptions to General Design Criteria 17, as discussed in Section 3.1, are due to the AP1000 design not requiring ac power sources for a design basis accident. Subsection 8.2.2.5 provides further discussion.

B-56 Diesel Reliability

Discussion:

Generic Safety Issue B-56 addresses the reliability of emergency onsite diesel-generators. Diesel reliability is a factor in the criteria associated with the resolution of Unresolved Safety Issue A-44. The resolution of issue B-56 is the development of guidelines for an acceptable emergency diesel-generator reliability program to ensure conformance with the emergency diesel-generator target reliability (0.95 to 0.975) identified in the proposed resolution of Unresolved Safety Issue A-44.

AP1000 Response:

The AP1000 diesel-generators are not safety related. The AP1000 diesel-generator reliability is based on diesel-generator industry standards and practices. The diesel generator is discussed in subsection 8.3.1. The diesel generator reliability is modeled in the PRA. The reliability assurance program is discussed in Section 16.2.

B-61 Allowable ECCS Equipment Outage Periods

Discussion:

Generic Safety Issue B-61 addresses surveillance test intervals and allowable equipment outage periods in the technical specifications for safety-related systems. This task involves the NRC development of analytically based criteria for use in confirming or modifying these surveillance intervals and allowable equipment outage periods.

AP1000 Response:

The AP1000 surveillance test intervals and allowable outage times help to meet plant safety goals while maximizing plant availability and operability. In determining these limits for the AP1000 technical specifications, a combination of NUREG-1431 precedent, system design, and safety-related function is considered.

B-63 Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Discussion:

Generic Issue B-63 addresses the adequacy of the isolation of low-pressure systems that are connected to the reactor coolant pressure boundary. The NRC staff requires that valves forming the interface between high- and low-pressure systems associated with the reactor coolant boundary have sufficient redundancy to prevent the low-pressure systems from being subjected to pressures that exceed their design limits.

AP1000 Response:

The AP1000 includes interconnections between high- and low-pressure systems. Each of these systems interfaces contains appropriate isolation provisions. Valves at the interface between high- and low-pressure systems have redundancy to prevent low-pressure systems from being subjected to pressures that exceed their design limits. The AP1000 design meets the provisions of the Standard Review Plan, Section 3.9.6.

The normal residual heat removal system interface is addressed in subsection 5.4.7. WCAP-15993 (Reference 56) provides an evaluation of the AP1000 conformance to intersystem loss-of-coolant accident regulatory criteria.

B-66 Control Room Infiltration Measurements

Discussion:

Generic Safety Issue B-66 addresses the adequacy of control room area ventilation systems and control building layout to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases. The NRC considers this issue as being technically resolved, and criteria have been incorporated in Standard Review Plan, Section 6.4.

AP1000 Response:

The AP1000 main control room is essentially leak-tight. A description of the control room habitability systems is contained in Section 6.4.

Verification of design infiltration rates is as specified in Standard Review Plan, Section 6.4. The AP1000 minimizes unfiltered in-leakage by maintaining the main control room at a slightly positive pressure.

C-1 Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Discussion:

Generic Issue C-1 addresses the long-term capability of hermetically sealed instruments and equipment that must function in post-accident environments. The NRC considers this issue as being technically resolved with the issuance of current criteria for qualification of safety-related electrical equipment.

AP1000 Response:

The AP1000 environmental qualification program described in response to Unresolved Safety Issue A-24 addresses qualification of safety-related instrumentation and electrical equipment that must function under accident conditions. This program confirms the integrity of seals employed in the design of Class 1E equipment. See item A-24 of this subsection and Section 3.11 for AP1000 qualification methodology.

C-4 Statistical Methods for ECCS Analysis

Discussion:

Generic Issue C-4 addresses NRC development of a statistical assessment of the certainty level of the peak clad temperature limit. Appendix K, "ECCS Evaluation Models," to 10 CFR 50 specifies the requirements for ECCS analysis. These requirements call for conservatisms to be applied to certain models and assumptions used in the analysis to account for data uncertainties at the time Appendix K was written. The resulting conservatism in the calculated peak clad temperature (PCT) has not been thoroughly compared against the uncertainty in peak clad temperature obtained from a realistically calculated (best-estimate) LOCA. The staff allows voluntary use of statistical uncertainty analysis to justify relaxation of all but the required conservatisms contained in current ECCS evaluation models.

AP1000 Response:

Chapter 15 discusses the LOCA analysis for the AP1000.

C-5 Decay Heat Update

Discussion:

Generic Issue C-5 involves following the work of research groups in determining best-estimate decay heat data and associated uncertainties for use in LOCA calculations.

The staff has determined that the 1979 ANSI 5.1 is technically acceptable and has allowed the use of this data to justify relaxation of non-required conservatisms in current ECCS evaluation models. The ECCS rule change allows the use of this new data. This issue was determined to be resolved.

AP1000 Response:

The large-break LOCA analyses for the AP1000, which employ the best-estimate W COBRA/TRAC analysis methodology (subsection 15.6.5), use the decay heat model identified in the 1979 ANSI 5.1 (Reference 26).

C-6 LOCA Heat Sources

Discussion:

Generic Issue C-6 addresses the impact on LOCA calculations of LOCA heat sources, their associated uncertainties, and the manner in which they are combined. An evaluation was made of the combined effect of power density, decay heat, stored energy, fission power decay, and their associated uncertainties with regard to calculations of LOCA heat sources.

AP1000 Response:

See subsection 15.6.5 for a discussion of LOCA heat sources.

C-10 Effective Operation of Containment Sprays in a LOCA

Discussion:

Generic Issue C-10 addresses the effectiveness of containment sprays to remove airborne radioactive materials that could be present within the containment following a LOCA. The NRC considers this issue as being technically resolved with the issuance of ANSI 56.5-1979 (Reference 28), which is referenced in Standard Review Plan, Section 6.5.2.

AP1000 Response:

The AP1000 design does not employ a safety-related containment spray system for removal of airborne radioactive materials in containment. Subsection 15.6.5.3 provides details of source term and mitigation techniques.

C-17 Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Discussion:

Generic Issue C-17 addresses the development of criteria for acceptability of radwaste solidification agents. The NRC considers this issue as technically resolved with the issuance of 10 CFR 61.56.

AP1000 Response:

The AP1000 solid radwaste system transfers, stores, and prepares spent ion exchange resins for disposal. It also provides for disposal of filter elements; sorting, shredding, and compaction of compressible dry active wastes. The solid radwaste system does not provide for liquid waste concentration or solidification. These functions, if used, are provided using mobile systems. Solidification of wastes is not performed by permanently installed systems.

1.9.4.2.3 New Generic Issues

These items were identified in NUREG-0933 as New Generic Issues and surfaced after the publication of the NUREGs that included the Task Action Plan items other unresolved safety issues.

Issue 14 PWR Pipe Cracks

Discussion:

This issue addresses the occurrences of main feedwater line cracking found in operating plants. This issue was classified as resolved with no new requirements.

AP1000 Response:

The design and inspection requirements for the feedwater lines are discussed in subsection 10.4.7.

Issue 15 Radiation Effects on Reactor Vessel Supports

Discussion:

Generic Safety Issue 15 addresses the potential problem of radiation embrittlement of reactor vessel support structures. There is a potential for radiation embrittlement of the reactor vessel support structure from long-term exposure to neutrons with an energy of 1 MeV or greater. Embrittlement due to neutron damage may increase the potential for propagation of existing flaws.

AP1000 Response:

The supports for the AP1000 reactor pressure vessel are designed for loading conditions and environmental factors including consideration of neutron fluence levels. The material requirements include fracture toughness requirements and impact testing requirements in compliance with ASME Code, Section III, Subsection NF requirements. The reactor pressure vessel supports are not in the region of high neutron fluence where neutron embrittlement of the supports would be a significant concern.

Issue 22 Inadvertent Boron Dilution Event

Discussion:

Some operating plants do not have provisions to detect boron dilution during cold shutdown. This could result in inadvertent criticality. The NRC staff concluded that existing review criteria are adequate. This issue was classified as resolved with no new requirements.

AP1000 Response:

The provisions in the design to preclude inadvertent boron dilution events are outlined in DCD subsection 9.3.6.

Issue 23 Reactor Coolant Pump Seal Failures

Discussion:

Generic Safety Issue 23 addresses reactor coolant pump seal failures that challenge the makeup capacity in PWRs. Such seal failures represent small-break loss-of-coolant accidents.

AP1000 Response:

The AP1000 reactor coolant pumps are sealless pumps. A sealless pump contains the motor and all rotating components inside a pressure vessel designed for full reactor coolant system pressure. The shaft for the impeller and rotor is contained within the pressure boundary; therefore, seals are not required in order to restrict leakage out of the pump into containment. Subsection 5.4.1 provides additional information on the sealless pump design for the AP1000 reactor coolant pumps. Since the reactor coolant pumps do not rely on seals as a reactor coolant pressure boundary, this issue is not applicable to the AP1000.

Issue 24 Automatic ECCS Switchover to Recirculation

Discussion:

This issue addresses the issue of switchover from safety injection to recirculation using manual valve alignment or automatic valve alignment.

AP1000 Response:

The AP1000 does not switch from injection to recirculation in the sense that injection is not isolated when recirculation is opened. The AP1000 does provide for automatic opening of the recirculation line on a low level signal from the in-containment refueling water storage tank. See Section 6.3 for additional details.

Issue 29 Bolting Degradation or Failure in Nuclear Power Plants

Discussion:

Generic Safety Issue 29 addresses a concern about pressure boundary integrity and component support reliability associated with bolt failures.

As documented in Generic Letter 91-17, the NRC has provided resolution of this issue. The resolution is documented in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," and NUREG-1445, "Regulatory Analysis for the Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." The resolution was based on a number of industry initiatives and NRC staff actions. NRC staff actions include issuing a number of bolting-related bulletins, generic letters and information notices. Industry initiatives include the publishing of EPRI Reports NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and NP-5067, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel."

EPRI Report NP-5769 establishes the characteristic that bolted connections exhibit leakage prior to failure resulting from bolt degradation. The NRC has endorsed the recommendation in NP-5769 that plant-specific bolting integrity programs be established that encompass safety-related bolting. NUREG-1339 includes recommendations and guidelines for the content of a comprehensive bolting integrity program.

AP1000 Response:

The elements of resolution pertain to the design, material selection, fabrication, and in-service inspection of the bolted connections found in the AP1000. To address this, resolutions found in NUREG-1339 are incorporated into the design, material selection, fabrication, and maintenance of the bolted connections. The maintenance practices are addressed by the maintenance program of the combined license holder. Conformance to ASME Code, Section III requirements for pressure boundary components and related supports provides safe operation in the event of bolting degradation. Because of the emphasis in the AP1000 design on access for maintenance and inspection, the recommended maintenance practices can be implemented.

Issue 43 Reliability of Air Systems

Discussion:

This issue addresses the concern that compressed air system degradation or malfunction may cause malfunction of safety-related systems and components. Of particular interest are air operated valves because of problems with the quality of the air supply or the manner in which the compressed air system fails. Generic Letter 88-14 and NUREG-1275 were issued in response to this issue.

AP1000 Response:

The compressed air systems are described in subsection 9.3.1. Provisions are included to maintain the quality of the air supply. The AP1000 safety-related, air-operated valves do not rely on the air supply to perform their safety-related function.

Issue 45 Inoperability of Instrumentation Due to Extreme Cold Weather

Discussion:

Generic Safety Issue 45 addresses the inoperability of instrumentation due to extreme cold weather. This issue was resolved with the issuance of changes to Standard Review Plan, Section 7.1, Appendix A to Section 7.1, Section 7.5, and Section 7.7.

AP1000 Response:

The AP1000 complies with Standard Review Plan Section 7.1, Appendix A to Section 7.1, Section 7.5, and Section 7.7.

Issue 51 Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Discussion:

Generic Safety Issue 51 addresses the susceptibility of open cycle service water systems to fouling including the buildup of aquatic bivalves and corrosion products that can significantly degrade the performance of the system. In operating plants, the service water system is typically used to cool safety-related equipment and to transfer decay heat to the ultimate heat sink.

AP1000 Response:

The service water system in the AP1000 provides cooling water to the component cooling water system and has no safety-related functions. None of the safety-related equipment requires cooling water to effect a safe shutdown or mitigate the effects of design basis events. Heat transfer to the ultimate heat sink is accomplished by heat transfer through the containment shell to air and water flowing on the outside of the shell.

The design of the service water system and the provisions for minimizing long-tern corrosion and organic fouling are described in subsection 9.2.1.

Issue 57 Effects of Fire Protection System Actuation

Discussion:

Generic Safety Issue 57 addresses the potential for adverse interactions from actuation of the fire protection system with safety-related equipment. Operating experience has shown that safety-related equipment subject to fire protection system water spray and other suppressant chemicals can be rendered inoperable.

AP1000 Response:

The fire protection system and fire protection program in the AP1000 minimize the potential for adverse interactions of safety-related equipment with the fire protection system. The means used to achieve this result include: isolating combustible material and limiting the spread of fire by subdividing the plant into fire areas separated by fire barriers, providing separate and redundant safe shut down components and associated electrical divisions to preserve the ability to safely shutdown the plant following a fire, and providing floor drains sized to remove expected firefighting water without flooding safety-related equipment. The design of the fire protection system is described in subsection 9.5.1.

Issue 67.3.3 Improved Accident Monitoring

Discussion:

This issue addresses weaknesses in accident monitoring. The recommended solution is to implement Regulatory Guide 1.97.

AP1000 Response:

The guidance of Regulatory Guide 1.97 is followed to determine the appropriate parameters to monitor in the AP1000.

Issue 73 Detached Thermal Sleeves

Discussion:

This issue addresses problems with "generation 3" thermal sleeves.

AP1000 Response:

The AP1000 does not use generation 3 thermal sleeves and includes design provisions to preclude failures of thermal sleeves.

Issue 75 Generic Implications of ATWS Events at the Salem Nuclear Plant

Discussion:

This issue considers the failure of reactor trip breakers to open and issues related to design and testing of the reactor protection system. Issues to be considered include the capability to record and display reactor trip system parameters, equipment classification information, post-maintenance testing, and reliability improvements in operating plants. Generic letter 83-28 and IE Bulletins 83-01 and 83-04 were issued by the staff with specific requirements.

AP1000 Response

The design of the reactor trip breakers and the reactor protection system is outlined in Section 7.1. Information on the functional requirements for reactor trip and conformance with industry and regulatory guidance is outlined in Section 7.2.

The provisions provided to display and record parameters used by the reactor trip system are outlined in subsections 7.1.2.6 and 7.1.2.13. Section 7.5 also provides information on requirements for safety-related display information.

Subsection 7.1.1 identifies the safety-related functions provided by the protection and safety monitoring system and the items that are included in the system including the reactor trip switchgear. Conformance of safety-related systems and components to industry and regulatory criteria is identified in subsection 7.1.4.

The reliability and fault tolerance of the protection and safety monitoring system for test maintenance and bypass conditions are outlined in subsection 7.1.2.10.

The changes in the design of the reactor trip breakers and associated logic to enhance reliability in operating nuclear power plants have been incorporated in the AP1000 design as appropriate. The reactor trip system includes built-in test capability.

WCAP-15800 addresses conformance with generic letters and bulletins.

Issue 79 Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

Discussion:

Generic Safety Issue 79 addresses the thermal stresses that occur in the reactor vessel head flange during a natural circulation cooldown. High stresses in the flange or studs during a natural circulation cooldown in PWRs could violate ASME code allowables. Cycling of the stresses could reduce the fatigue margin. Generic Letter 92-02 repeated the reporting requirements of 10CFR 50.73 (a)(2)(ii)(B), "Licensee event report system."

AP1000 Response:

The natural circulation cooldown transient is evaluated as part of ASME Code vessel evaluations and is discussed in Subsection 3.9.1.1.2.11. The reporting requirements to address the requirements of 10CFR 50.73 (a)(2)(ii)(B) referenced in Generic Letter 92-02 are the responsibility of the Combined License holder.

Issue 82 Beyond Design Basis Accidents in Spent Fuel Pools

Discussion:

This issue addresses the concern of a beyond design basis accident in which the spent fuel pool is drained and spent fuel stored there subsequently catches on fire releasing very large amounts of radioactive contamination. This issue is classified as resolved with no new requirements.

AP1000 Response:

The AP1000 includes design provisions that preclude draining of the spent fuel pool. Also, provisions are available to supply water to the pool in the event the water covering the spent fuel begins to boil off.

Issue 83 Control Room Habitability

Discussion:

Loss of control room habitability following an accidental release of external toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor. Use of the remote shutdown workstation outside the control room following such events is unreliable since this station has no emergency habitability or radiation protection provisions.

AP1000 Response:

Habitability of the main control room is provided by the main control room/control support area HVAC subsystem of the nonsafety-related nuclear island nonradioactive ventilation system (VBS). If ac power is unavailable for more than 10 minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The safety-related main control room emergency habitability system supplies breathable quality air for the main control room operators while the main control room is isolated.

In the event of external smoke or radiation release, the nonsafety-related nuclear island nonradioactive ventilation system provides for a supplemental filtration mode of operation, as discussed in Section 9.4. In the unlikely event of a toxic chemical release, the safety-related main control room emergency habitability system has the capability to be manually actuated by the operators. Further, a 6-hour supply of self-contained portable breathing equipment is stored inside the main control room pressure boundary.

Issue 87 Failure of HPCI Steam Line Without Isolation

Discussion:

Generic Safety Issue 87 addresses the uncertainty regarding the operability of the motor-operated isolation valves for the steam supply lines of the high-pressure coolant injection (HPCI) system in boiling water reactors following a postulated break in the supply line. A break in the line could lead to high flow or high differential pressure that may inhibit closure of the isolation valve. These valves typically cannot be tested in-situ for the design flow rates and pressures. Although the AP1000 does not have a high-pressure coolant injection system, it does have isolation valves designed to close against high flow or high pressure differential in the event of a postulated pipe break.

The issue of the operability of motor-operated valves has received considerable attention since Generic Safety Issue 87 was initiated. The NRC provided guidance for inservice testing of motor-operated, safety-related valves in Generic Letter 89-10. SECY-93-087 identifies the proposed position on inservice testing of safety-related valves for advance light water reactors. The guidance in these documents recommends that safety-related valves be tested under full flow under actual plant conditions where practical. EPRI has a program to demonstrate operation of motor-operated valves.

AP1000 Response:

Safety-related valves must meet the requirements of ASME Code, Section III to provide pressure boundary integrity. Valves and valve operators are sized to provide operation under a full range of design basis flow and pressure drop conditions. For the AP1000, safety-related motor-operated valve designs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against maximum pressure differential and flow. The requirements for this testing are based on ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants." See subsection 5.4.8 for an outline of AP1000 valve requirements.

The in-service testing program for safety-related valves is discussed in subsection 3.9.6. Motor-operated valves are to be operability tested as outlined in subsection 3.9.6.2.2. Subsection 3.9.6.2.2 includes a discussion of the factors to be considered to determine which valves and test conditions are to be used for operability testing of power-operated valves. Sufficient flow is provided to fully open check valves during testing unless the maximum accident flows are not sufficient to fully open the check valve. The valves built to ASME Code, Section III are tested in compliance with the requirements found in the ASME code, "Code for Operation and Maintenance of Nuclear Power Plants." For additional information on inservice testing of safety-related valves, see subsection 3.9.6.

Issue 93 Steam Binding of Auxiliary Feedwater Pumps

Discussion:

Generic Safety Issue 93 addresses the potential for a common mode failure of the pumps in an auxiliary or emergency feedwater system. Hot water leaking through one or more isolation valves can flash to steam at the auxiliary feedwater pump potentially resulting in the failure of the pump to operate if required because of steam binding. The NRC addressed this issue in Bulletin 85-01, and reinforced it in Generic Letter 88-03, by requesting that the fluid conditions in the auxiliary feedwater system be monitored and procedures be developed to recognize steam binding and restore the auxiliary feedwater system to operable status if steam binding should occur.

AP1000 Response:

The AP1000 does not have a safety-related auxiliary feedwater system. The passive core cooling system provides the safety-related function of cooling the reactor coolant system in the event of loss of feedwater. The startup feedwater system provides the steam generators with feedwater during plant conditions of startup, hot standby, and cooldown and when the main feedwater pumps are unavailable. The startup feedwater system has no safety-related function.

The startup feedwater system includes temperature instrumentation in the pump discharge for monitoring of the temperature of the startup feedwater system. The system also includes a normally closed isolation valve and a normally closed check valve for each pump limiting potential back leakage.

Issue 94 Additional Low-Temperature Overpressure Protection for Light Water Reactors

Discussion:

Generic Safety Issue 94 addresses the establishment of additional guidance for reactor coolant system low-temperature overpressure protection to ensure reactor vessel and reactor coolant system integrity beyond that identified in the resolution to Generic Safety Issue (GSI) A-26. Low-pressure overpressurization events that occurred subsequent to the implementation of the guidelines for resolution of GSI A-26 indicated a need for additional low-temperature overpressure protection. To resolve this issue, the NRC issued Generic Letter 90-06 which required a revision to plant technical specifications for operability of the low-temperature overpressure protection system. Other possible solutions identified in GL 90-06 included hardware modifications including use of residual heat removal system relief valves and requiring the low temperature overpressure protection system to be fully safety related.

AP1000 Response:

The reactor vessel for the AP1000 is designed to be less susceptible to brittle fracture during low temperature overpressure events. The material requirements and welding processes are developed to enhance resistance to embrittlement. See subsection 5.3.2 for additional information on the requirements to address fracture toughness of the reactor vessel.

The normal residual heat removal system is designed to provide the safety-related function of low temperature overpressure protection for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure within the limits specified in 10 CFR 50, Appendix G. The relief valve in the normal residual heat removal system is used to provide the overpressure protection. See subsection 5.4.7 for additional information on the design of the normal residual heat removal system and the overpressure protection function.

Issue 103 Design for Probable Maximum Precipitation

Discussion:

Generic Safety Issue 103 addresses the methodology used for determining the design flood level for a particular reactor site. This issue was resolved by incorporating the methodology into the Standard Review Plan.

AP1000 Response:

This is a site-related parameter. The AP1000 is designed for air temperatures, humidity, precipitation, snow, wind, and tornado conditions as specified in Table 2-1. The site is acceptable if the site characteristics fall within the AP1000 plant site design parameters in Table 2-1. For cases where a site characteristic exceeds the envelope parameter, see Chapter 2.

Issue 105 Interfacing System LOCA at BWRs

Discussion:

Generic Safety Issue 105 addresses concerns over the adequacy of isolation valves between the reactor coolant system and low-pressure interfacing systems in BWRs. This issue, which is limited to pressure isolation valves in BWRs, is related to Generic Safety Issue 96, which considers the failure of the pressure isolation valves between the reactor coolant system and the RHR system in PWRs. Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release. Designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue.

AP1000 Response:

For information on this issue, see subsection 1.9.5.1, SECY-90-016 Issues. See subsection 5.4.7 for additional information on the normal residual heat removal system design.

Issue 106 Piping and Use of Highly Combustible Gases in Vital Areas

Discussion:

Generic Safety Issue 106 addresses the normal process system use of relatively small amounts of combustible gases on site and also addresses leaks or breaks in the hydrogen piping and supply system that could result in the accumulation of a combustible or an explosive mixture of air and hydrogen within the auxiliary systems building. The accumulation of combustible or explosive mixtures of gas in the auxiliary systems building could represent a threat to safety-related equipment if the combustible gases are inadvertently ignited.

AP1000 Response:

The AP1000 uses small amounts of combustible gases for normal plant operation. Most of these gases are used in limited quantities and are associated with plant functions or activities that do not jeopardize any safety-related equipment. These gases are found in areas of the plant that are removed from the Nuclear Island (see subsection 9.3.2 for a description of the plant gas system). The exception to this is the hydrogen supply line to the chemical and volume control system (CVS).

The chemical and volume control system is the only system on the nuclear island that uses hydrogen gas. Hydrogen is supplied to the AP1000 CVS inside containment from a single hydrogen bottle. The release of the contents of an entire bottle of hydrogen in the most limiting building volumes (both inside containment and in the auxiliary building) would not result in a volume percent of hydrogen large enough to reach a detonable level.

The chemical and volume control system hydrogen supply piping is routed through the turbine building and into the auxiliary building and then into containment. The H_2 supply line is routed through the piping/valve room on elevation 100′-0″ of the auxiliary building. The piping/valve penetration room in the auxiliary building on elevation 100′-0″ is designed as a 3-hour fire zone. A fire in this area would not inhibit the safe shutdown of the plant. More information is contained in Appendix 9A.

The turbine building does not house any safety-related systems or equipment. The release of hydrogen into an area of the turbine building does not represent a threat to the safety of the plant.

The AP1000 containment has hydrogen sensors that would detect hydrogen leaks. The containment hydrogen concentration monitoring subsystem is described in Subsection 6.2.4.1.

Issue 113 Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

Discussion:

Generic Safety Issue 113 addresses the requirements for qualification and periodic operability testing of large bore hydraulic snubber for operating plants. Large-bore hydraulic snubbers are used to a limited extent on the AP1000 to provide support, particularly for seismic events, of piping systems and components while allowing for movement due to thermal expansion. The NRC, in a draft regulatory guide (SC-708-4, "Qualification and Acceptance Test for Snubbers Used in Systems Important to Safety"), has established recommendations for testing of hydraulic snubbers on a forward-fit basis; that is, units without a license at the time the recommendations were established.

AP1000 Response:

The AP1000 plant uses significantly fewer hydraulic snubbers than do currently operating plants. In addition to the recommendations in the draft regulatory guide, testing requirements have been established in ASME OM Code – 1995 Edition up to and including the 1996 Addenda, "Code for Operation and Maintenance of Nuclear Power Plants." Subsection 3.9.3.4.3 discusses requirements for production and qualification testing. The design of the hydraulic snubbers permits required preoperational and inservice testing.

Subsection 3.9.8.3 defines the responsibility to provide information on snubber operability testing.

Issue 120 On-Line Testability of Protection System

Discussion:

This issue is related to the protection system of some older plants that do not provide for as complete a degree of on-line protection system testing surveillance capability as is now required. Testing requirements and guidance are found in GDC 21, Regulatory Guides 1.22 and 1.118 and IEEE Standard 338. This item is classified as resolved with no additional requirements.

AP1000 Response:

This item does not apply to the AP1000. The provision for testing of the protection system in conformance with the regulatory guidance is found in Section 7.1.

Issue 121 Hydrogen Control for Large, Dry PWR Containments

Discussion:

Generic Safety Issue 121 concerns ongoing NRC experimental and analytical programs addressing the likelihood of safe shutdown equipment surviving a hydrogen burn. The staff also intends to explore the possibility and probable consequences of the formation of local detonable concentrations in large, dry PWRs. The concerns are prediction of conditions in realistic configurations, and containment and equipment survivability.

AP1000 Response:

The AP1000 includes provisions for hydrogen control for the unlikely severe accident cases in which large amounts of hydrogen could be generated because of degraded core events. Analyses were performed to examine the consequences of hydrogen burn and to evaluate the likelihood of deflagration to detonable transitions.

For severe accident cases, the containment hydrogen control system prevents hydrogen burn initiation at high hydrogen concentration levels. Hydrogen igniters promote burning when the lower flammability limit is reached and limits the containment hydrogen concentration to less than 10 volume percent during and following a degraded core or core melt.

Thus, for severe accident cases, the AP1000 is designed to prevent the occurrence of hydrogen detonation, thereby preventing the possibility of the resultant large pressure spikes in containment, which is the source of concern for containment integrity and equipment survival. Details of the hydrogen ignition subsystem are provided in subsection 6.2.4.2.3. Placement of the hydrogen igniters is discussed in subsection 6.2.4.

A hydrogen burn analysis shows that the AP1000 hydrogen igniter system is effective in maintaining the hydrogen concentration throughout the containment close to the lower flammability limit, and that the peak pressure in the containment during and following hydrogen burn remains well below ASME service level C stress intensity limits. The hydrogen concentration is similar in all compartments analyzed, indicating that the hydrogen released mixes well in the AP1000 containment. The analyses predict conditions in realistic configurations. Peak gas temperatures and pressures in each compartment for each case analyzed are provided, thus providing the hydrogen burn thermal environment that containment equipment will experience. Details are provided in Chapter 14 of the PRA report.

The challenge to the AP1000 containment integrity from hydrogen deflagrations and detonations during core damage events is examined in the hydrogen deflagration and detonation analyses. This bounding evaluation assumes that an amount of hydrogen equivalent to 100-percent active cladding oxidation burns all at once in the AP1000 containment, with no credit taken for the hydrogen igniters. The evaluation concludes that a hydrogen deflagration is unlikely to cause containment failure. Other analyses show that a deflagration to detonation transition in any part of the AP1000 containment is unlikely. Containment failure from a detonation is not considered a credible event for the AP1000 because of the lack of conditions supporting a deflagration to detonation transition, the provision and placement of hydrogen igniters, and the containment design features resulting in a well-mixed atmosphere. Details are provided in subsection 10.2.5 of the PRA evaluation report.

The hydrogen igniters and the containment electrical and mechanical penetrations are designed to operate in the most limiting severe accident environment, including a hydrogen burn. (See subsection 10.2.5 of the PRA evaluation report.) The approach of using controlled burning to prevent accidental hydrogen burn initiation provides confidence that safety-related equipment will continue to operate during and after hydrogen burns. (See subsection 6.2.4.)

Issue 124 Auxiliary Feedwater System Reliability

Discussion:

Generic Safety Issue 124 addresses the use of probabilistic risk assessment to evaluate the reliability of the auxiliary feedwater system. The issue was resolved by the NRC's issuing plant-specific requirements for a few plants that did not initially have a reliability higher than a minimum criteria.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not have a safety-related auxiliary feedwater system. The passive core cooling system provides the safety-related function of cooling of the reactor coolant system in the event of loss of feedwater. The startup feedwater system provides the steam generators with feedwater during plant conditions of startup, hot standby, and cooldown and when the main feedwater pumps are unavailable. The startup feedwater system has no safety-related function beyond containment isolation.

Issue 128 Electrical Power Reliability

Discussion:

Generic Safety Issue 128 addresses the reliability of onsite electrical systems and encompasses GSI 48, GSI 49, and GSI A-30.

AP1000 Response:

The design basis and design criteria for the Class 1E dc and UPS system is provided in subsections 8.1.4.2.1 and 8.1.4.3. The class 1E dc and UPS system design is described in subsection 8.3.2.1.1. Specifically, this design addresses IEEE Standards 603 and 308. This includes the following generic issues:

- Generic Safety Issue 48, LCO for Class 1E vital instrument buses in operating reactors. Chapter 16 provides the AP1000 technical specifications. Subsections 16.1.3.8.3 and 16.1.3.8.4 provide the limiting conditions for operation in the event of a loss of one or more

Class 1E 120-vac vital instrument buses and the associated inverters. The AP1000 Class 1E buses have no tie breakers

- Generic Safety Issue 49, interlocks and LCOs for Class 1E tie breakers. Based on the historical background, this issue is not applicable to the AP1000 design. There are no tie breakers between the four class 1E divisions.

- Generic Safety Issue A-30, adequacy of safety-related dc power supplies. The AP1000 incorporates the following recommended enhancements:

  – The Class 1E dc distribution system design is in accordance with the guidelines of IEEE Standard 384 and Regulatory Guide 1.75.

  – Four separate divisions of Class 1E dc power are provided.

The AP1000 design provides additional testing capability through the installed spare battery bank with one installed battery charger. The spare battery bank permits frequent full-component testing without compromising plant availability. Battery equalization can be performed off-line. The battery and battery charger can be tested and maintained separately.

Issue 130 Essential Service Water Pump Failure at Multiple Plant Sites

Discussion:

Generic Safety Issue 130 addresses the use of shared or cross-connected essential service water systems at sites with two or more reactor plants. During some situations the crosstied pumps may not be available for accident mitigation operations.

AP1000 Response:

The AP1000 is a single, independent plant that does not share or cross-tie systems or components with another plant. See Section 1.2 for a general description of the plant. This issue is not applicable to the AP1000.

Issue 135 Integrated Steam Generator Issues

Discussion:

Generic Safety Issue 135 was initiated to provide an integrated work plan for the resolution of steam generator issues including steam generator overfill consequences, water hammer, and eddy current testing. The issue was divided into the following four tasks:

1. Assessing current capabilities of eddy current testing and developing recommendations.

2. Reviewing SGTR results and conclusions to develop regulatory analysis supporting Standard Review Plan changes.

3. Reassessing SGTR associated issues including radiological, design basis, tube integrity, procedures, and RCS pressure control.

4. Reviewing the effects of water hammer, overfill and water carryover.

The results of the tasks will provide the staff with a basis to develop a position on offsite dose, operator action, tube integrity, water hammer, and valve operability.

AP1000 Response:

The AP1000 design features are discussed below.

TASK 1: Appendix 1A identifies the level of conformance with Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." As detailed in Appendix 1A, the AP1000 conforms with the regulatory guidance except where state-of-the-art advances have enhanced inservice inspection techniques. Further, as specified in subsection 5.4.2.5, the steam generators permit access to tubes for inspection and/or repair or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. The AP1000 steam generator includes features to enhance robotics inspection of steam generator tubes without manned entry of the channel head.

TASK 2: Subsection 15.6.3.1.4 discusses anticipated operator recovery actions and the effects of those actions in the mitigation of a steam generator tube rupture (SGTR). As discussed in subsection 15.6.3.2, the AP1000 incorporates automatic steam generator overfill protection. The details of the design are provided in subsection 15.6.3.2, with the control logic provided in Section 7.2.

TASK 3: The following sections of the DCD provide pertinent details on SGTR issues.

- Reassessment of radiological consequences: Subsection 15.6.3 provides details of the scenario, analysis assumptions, and results.

- Re-evaluation of design basis SGTR: The design basis SGTR evaluated on the AP1000 is discussed in subsection 15.6.3, providing details of the scenario, analysis assumptions and results.

- Supplemental Tube Inspections: See subsection 5.4.2.5, Appendix 1A, Regulatory Guide 1.83.

- Denting criteria: Subsection 5.4.2.4.3 provides a discussion of steam generator design and tubing compatibility with secondary coolants.

- Improved accident monitoring and reactor vessel inventory measurement: Section 7.5 discusses the safety related display information.

- Reactor coolant pump trip: Subsection 7.3.1.2.5 discusses reactor coolant pump trip.

- Control room design: Sections 7.5 and 18.8 discuss the control room design and design process.

- Emergency operating procedures: Subsection 18.9 addresses the development of emergency operating procedures.

- Organizational responses: Chapter 13 identifies the requirements for organizational responses.

- Reactor coolant pressure control: Subsection 7.7.1.6 addresses primary system pressure control.

TASK 4: Steam generator overfill, water carryover and water hammer are addressed as discussed in subsection 15.6.3.2, with the control logic provided in Section 7.2.

Issue 142 Leakage Through Electrical Isolators in Instrumentation Circuits

Discussion:

Generic Issue 142 addresses the susceptibility to leakage of isolation devices between safety- and nonsafety-related electrical systems. The NRC requires that licensees identify isolation devices in instrumentation circuits that are potentially susceptible to electrical leakage, define and perform an inspection and test program, replace failed or unacceptable isolators, and implement an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

AP1000 Response:

The use of isolation devices in the AP1000 Instrumentation and Control Architecture is described in subsections 7.1.2.10, "Isolation Devices," 7.7.1.11, "Diverse Actuation System," and WCAP-15776 (Reference 70), Section 3.9, “Conformance to the Requirements to Maintain Independence Between Safety Systems and Other Interconnected Equipment (Paragraph 5.6.3.1 of IEEE 603-1991).” As stated in WCAP-15776, Section 3.9, the isolation devices are tested to conform to requirements. This testing meets the requirement for an inspection and test program and identifies those devices that are potentially susceptible to electrical leakage. Implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems is the responsibility of the Combined License holder. The use of fiber-optic data links eliminates electrically conductive paths between receiving and transmitting terminals, and eliminates the potential for electrically generated noise caused by leakage through an isolator. These communication links also use extensive testing and error checking to minimize erroneous transmissions. These data links are described in subsection 7.1.2.8, "Communication Functions." In addition, electromagnetic design, testing, and qualification is performed as described in WCAP-15776, Section 2.6, "Design Basis: Range of Conditions for Safety System Performance (Paragraph 4.7 of IEEE 603-1991.)"

Issue 143 Availability of Chilled Water System and Room Cooling

Discussion:

This issue relates to the need to maintain air cooling systems in some rooms containing safety-related system components.

AP1000 Response:

This issue does not apply to the AP1000. The AP1000 does not rely on active safety systems to provide safe shutdown of the plant. A total loss of HVAC systems will not prevent a safe shutdown.

Issue 153 Loss of Essential Service Water in LWRs

Discussion:

This issue is related to the reliability of essential service water and the failure of such systems due to fouling mechanisms, ice effects, design deficiencies, flooding, multiple equipment failures, and personnel errors. This issue has been the subject of a number of generic communications from the NRC staff.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not rely on the service water and component cooling water systems to provide safety-related safe shutdown.

Issue 163 Multiple Steam Generator Tube Leakage

Discussion:

This issue identifies a safety concern associated with potential multiple steam generator tube leaks triggered by a main steam line break outside containment that cannot be isolated. This sequence of events could lead to core damage due to the loss of all primary system coolant and safety injection fluid in the refueling water storage tank.

AP1000 Response:

The AP1000 plant response to a main steam line break (MSLB) scrams the reactor automatically and removes decay heat via the intact generator or the PRHR heat exchanger. If the MSLB is not isolated, the RCS will continue to lose coolant after shutdown through leaking steam generator tubes; the plant responds to the scenario as a small LOCA. The core makeup tanks drain and produce a low level signal. The plant protection and monitoring system depressurizes the RCS via the automatic depressurization system (ADS). The core remains covered throughout the scenario. Once the RCS is depressurized, the much lower reactor coolant system pressure stops the water loss through the leaking steam generator tubes. Therefore, no long-term core uncovery is expected.

Issue 168 Environmental Qualification of Electrical Equipment

Discussion:

This issue is related to the effects of cable aging and whether the licensing basis for older plants should be reassessed or enhanced in connection with license renewal, or whether they should be reassessed for the current license term.

AP1000 Position:

This issue applies to operating plants and does not apply to the AP1000.

Issue 185 Control of Recriticality Following Small-Break LOCAs in PWRs

Discussion:

This issue is related to the potential for large reactivity transients, including prompt criticality, and significant heat generation resulting from natural circulation flow of unborated water formed in steam generators following small-break LOCAs.

AP1000 Position:

This scenario is not a safety concern for the AP1000 because of the passive safety systems designed to mitigate the consequences of a LOCA. Specifically, the automatic depressurization system operates to reduce primary system pressure and, thus, prevents significant heat transfer in the steam generators. Consequently, the steam generators should not generate any significant amount of boron-free condensate via reflux condensation over an extended period during a LOCA event. In the AP1000 design, the steam generator functions as a "heat source" as the RCS depressurizes, rather than a "heat sink" as it does in conventional PWR designs. Therefore, the differential temperature across the primary and secondary side of the generators is such that steam from the reactor will not condense on the tubes.

Another important design feature of the AP1000 that reduces the significance of this event is the elimination of the loop seal in the inlet to the reactor coolant pump. By elimination of the crossover leg piping, a large volume of boron-free condensate cannot collect in the loop piping. Thus, restart of the reactor coolant pumps following a LOCA will not result in a large slug of unborated water entering the core.

Post-LOCA, the PRHR heat exchanger can act as a heat sink and potentially could be a source of unborated water post-LOCA. However, condensate from the PRHR heat exchanger outlet mixes with the borated injection from the core makeup tanks and accumulators, and adequately mixes in the reactor vessel downcomer to prevent post-LOCA boron dilution. Long-term boration of the core is provided by the injection from the borated IRWST.

Issue 191 Assessment of Debris Accumulation on PWR Sump Performance

Discussion:

This issue addresses new contributors to debris and possible blockage of PWR sumps. Generic Letter (GL) 2004-02 (Reference 2), issued in September 2004, identified actions that utilities must take to address the sump blockage issue. The NRC position is that plants must be able to demonstrate that debris transported to the sump screen after a LOCA will not lead to unacceptable head loss for the recirculating flow. For the AP1000, this requirement is interpreted as demonstrating that debris transported to recirculating screens will not significantly impede flow through the PXS and will not adversely affect the long-term operation of the PXS.

AP1000 Position:

The AP1000 Nuclear Power Plant uses natural recirculation for cooling the core following a loss of coolant accident (LOCA).

Screens are provided in strategic areas of the plant to remove debris that might migrate with the water in containment and adversely affect core cooling. Accordingly, it must be assured that the screens themselves are not susceptible to plugging.

Technical report APP-GW-GLR-079 (Reference 71) evaluates the potential for debris to plug the AP1000 screens consistent with Regulatory Guide 1.82 Revision 3 and subsequently issued Nuclear Regulatory Commission guidance. The evaluation considers the various potential contributors to screen plugging. It considers debris that could be produced by a LOCA as well as resident fibers and particles that could be present in containment prior to the LOCA. It considers the AP1000 containment design, equipment locations, and containment cleanliness program. The evaluation uses debris characteristics based on sample measurements from operating plants and evaluates the generation of chemical precipitants considering materials used inside the AP1000 containment, the post-accident water chemistry, and applicable research and testing. The AP1000 screen designs are acceptable.

1.9.4.2.4 Human Factors Issues

These issues were outlined in the Human Factors Program Plan and are documented in NUREG-0985, Revision 1. The Human Factors Program Plan includes the human factors tasks required to address NUREG-0660.

HF4.4 Guidelines for Upgrading Other Procedures

Discussion:

The need was evaluated to develop technical guidance for use in upgrading normal operating procedures and abnormal operating procedures, similar to what the NRC staff completed for emergency operating procedures. NUREG-0933 classified this item as resolved with no new requirements.

AP1000 Response:

The process to manage the development, review and approval of AP1000 Normal Operating, Abnormal Operating, Emergency Operating, Refueling and outage planning, Alarm response, Administrative, Maintenance, Inspection, Test and Surveillance Procedures as well as the procedures which address the operation of post-72 hour equipment is delineated in APP-GW-GLR-040 (Reference 72).

Writer's Guidelines have been developed which control the preparation of Normal Operating Procedures and Two-Column Format Procedures. The Writer's Guidelines establish programmatic guidelines. The criteria and methodology for procedure development is described in this technical report and in Westinghouse Writer's Guidelines, and Human Factors-related procedures have been developed in accordance with these criteria/guidelines.

HF5.1 Local Control Stations

Discussion:

Human Factors Issue 5.1 addresses the need to develop additional guidance for the design of local control stations.

AP1000 Response:

The AP1000 local control stations are designed using the same human factors engineering (HFE) design process as is used for the main control room (MCR). The human factors engineering design process is described in Chapter 18 of the DCD. Subsection 18.8 provides a description of the human system interface (HSI) design element of the overall design process. As part of the human system interface design process, design guidelines for each interface, such as workstation displays, are generated. These guidelines are used when designing the respective interface and control stations. This provides consistency of human system interface design, including local control stations, with the main control room.

HF5.2 Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

Discussion:

Human Factors Issue 5.2 addresses review criteria for human factors aspects of advanced controls and instrumentation.

AP1000 Response:

Chapter 18 of the DCD describes the human factors engineering (HFE) program for the AP1000. Section 18.4 includes a description of the Functional Requirements Analysis and Allocation (element 3) for the AP1000. The objective of this allocation process is to define the AP1000 safety function requirements and allocate functions between the human and the machine appropriately. Section 18.8 also presents the implementation plan for the human system interface (HSI) design. This description of the human system interface design process includes the development of design guidelines, the execution of man-in-the-loop concept testing, review of human system interface design, and the use of a full-scale mockup.

The AP1000 human system interface (HSI)/man-machine interface (MMI) includes the following resources:

- Alarm system
- Computerized Procedure System
- Plant Information System
- Qualified Data Process System (QDPS)
- Controls (dedicated and soft)
- Wall Panel Information System (WPIS)

The implementation plan for the design of each of these human system interfaces (HSI design) is described in section 18.8. The mission statements and high-level information for each of these resources is also provided in Section 18.8. The plant information system provides display at the operators workstation. The qualified data process system provides qualified (Class 1E) displays to operator, located at the dedicated safety panel. The alarm system provides alarm overviews which are integrated into the wall panel information system and it provides alarm support displays at the operator's workstation. Alarms are integrated into the workstation displays. There will be a navigational link from an alarm support display for a specific alarm to its associated alarm response procedure as presented to the operator by the computerized procedure system. Design guidelines for each human system interface is developed as part of the human system interface design (as described in subsection 18.8). These design guidelines are developed from existing industry guidelines and considerations specific to the technology planned for the human system interface. Human factors engineering specialists are part of the human factors engineering/ man-machine interphase design team (DCD Section 18.2) and will be involved in the development of the design guidelines.

1.9.5 Advanced Light Water Reactor Certification Issues

This subsection addresses the advanced light water reactor issues identified by the NRC in SECY-90-016 (Reference 29), in the February 27, 1992 NRC letter from D. M. Crutchfield to E. E. Kintner (Reference 30).

1.9.5.1 SECY-90-016 Issues

The following issues were outlined in SECY-90-016 (Reference 29).

1.9.5.1.1 Advanced Light Water Reactor Public Safety Goal

NRC Position:

Based on current regulatory guidance, including the NRC Severe Accident Policy Statement, Standardization Policy Statement, and Safety Goal Policy Statement, it is expected that any new standard plant design will result in a higher level of severe accident safety than current plant designs. This is achieved by improving safety and by striking a balance between accident prevention and mitigation.

The overall objective of the public safety goal is to significantly reduce or eliminate the likelihood of known major safety issues.

The safety goals approved by the NRC in the Staff Requirements Memorandum to SECY-90-016 (Reference 31) are as follows:

- The mean core damage frequency target for each design should be less than 1.0×10^{-4} per reactor year.

- The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation, where a large release is defined as one that has a potential for causing an early offsite fatality.

AP1000 Response:

The AP1000 level 1, 2, and 3 PRA evaluations for both internal and external events (excluding seismic events) demonstrate conformance with the NRC safety goals. The AP1000 PRA evaluates shutdown events and provides additional information and specific results.

1.9.5.1.2 Use of Physically-Based Source Term

NRC Position:

As noted in SECY-95-172 (Reference 57), the NRC plans to use the accident source term model from NUREG-1465 (Reference 58). This source term model provides a physically based approach to modelling of activity releases from the reactor core to the containment in the event of a core degradation accident. As discussed in SECY-94-302 (Reference 59), for the design basis accident, release of activity from the core will not be assumed to extend beyond the in-vessel release phase.

In calculating the radiological consequences of accidents, as stated in Reference 57, the NRC intends to use the model presented in SECY-94-194 (Reference 60) which identifies the proposed changes to 10 CFR Parts 50 and 100. The pertinent features that will be applied to the determination of accident radiological consequences are:

- In place of thyroid and whole body dose limits, dose limits are specified as total effective dose equivalent (TEDE). The offsite dose limits of 25 rem whole body and 300 rem thyroid are replaced by a limit of 25 rem TEDE. The dose limit for the control room operators (currently identified in SRP Section 6.4 as 5 rem whole body, 30 rem thyroid, and 30 rem beta skin) is replaced by 5 rem TEDE which is consistent with GDC 19.

- Instead of calculating the site boundary dose over the first two hours of the accident, the dose is to be calculated for the two hour interval over which the highest dose would be calculated.

AP1000 Response:

The AP1000 radiological consequence analysis utilizes the accident source term provided in Regulatory Guide 1.183.

1.9.5.1.3 Anticipated Transients Without Scram (ATWS)

NRC Position:

This former unresolved safety issue was resolved with the issuance of Rule 10 CFR 50.62. Requirements for currently operating pressurized water reactors include diverse reactor trip (except for Westinghouse plants) and diverse actuation of auxiliary feedwater and turbine trips.

The Staff Requirements Memorandum to SECY-90-016 (Reference 31) approved the requirement for diverse reactor trip systems for evolutionary advanced light water reactors. However, it added that if the applicant can demonstrate that the consequences of an anticipated transient without reactor trip are acceptable, the NRC should accept the demonstration as an alternative to the diverse reactor trip system.

AP1000 Response:

The AP1000 complies with the current rules on an anticipated transient without reactor trip as specified in 10 CFR 50.62.

The AP1000 design includes the following design features aimed at minimizing the probability of occurrence of an anticipated transient without reactor trip and at mitigating the consequences if it occurs.

- The design of the protection and safety monitoring system is highly reliable, using a two out of four coincidence logic and featuring continuous diagnostic testing. The system incorporates fail-safe features to the extent practical. It is designed to generate a reactor trip signal and to generate an actuation signal for most engineered safety features components when protection system failures occur.

- For a reactor trip, the switchgear consists of eight circuit breakers arranged in a two out of four matrix located in two separate cabinets. The trip is implemented by undervoltage trip attachments and diverse shunt trip devices on the circuit breakers. To initiate a reactor trip, power is interrupted to the undervoltage trip attachment, while the shunt trip attachment is energized. Either device trips the breaker. The eight-breaker configuration permits testing of the reactor trip breakers without the use of auxiliary bypass breakers.

- The reactor trip switchgear can be actuated manually from the main control room by reactor trip switches hard-wired to the shunt trip attachment and undervoltage coils for each reactor trip breaker. In addition, it is possible to manually initiate a reactor trip from the main control room by turning off the motor-generators that provide power for control rod operation.

- A nonsafety-related diverse actuation system is included in the AP1000 design. The diverse actuation system inserts control rods by de-energizing the field windings of the control rod motor-generators.

- The diverse actuation system trips the turbine and diversely actuates selected other engineered safeguards functions. Additional details of the diverse actuation system are included in Section 7.7.

Section 15.8 describes the evaluation of an anticipated transient without reactor trip.

1.9.5.1.4 Midloop Operation

NRC Position:

Loss of decay heat removal function has occurred on a number of occasions in operating plants. In response to these events, the NRC issued Generic Letter 87-12 requesting that operating plants provide information regarding mid-loop operation. Generic Letter 88-17 requested additional information and provided guidance to operating utilities. Subsequent NRC evaluations have indicated that loss of decay heat removal during midloop operation may contribute significantly to public risk.

It is the NRC position that for future plants, conformance with Generic Letter 88-17 is insufficient, and additional hardware features should be incorporated into the design.

The Staff Requirements Memorandum to SECY-90-016 (Reference 31) approved the proposed NRC position, with the following four additional recommendations made by the ACRS:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low liquid levels at the junction of the decay heat removal system suction lines and the reactor coolant system

- Provisions to ensure availability of reliable systems for decay heat removal

- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the decay heat removal system suction lines and the reactor coolant system

- Provisions for maintaining containment closure or for rapid closure of containment openings.

AP1000 Response:

The following features are incorporated into the design of the reactor coolant system and the normal residual heat removal system for continued performance of the residual heat removal function during midloop operation:

- The layout of the reactor coolant system hot leg piping and the steam generator channel head is such that installation of the nozzle dams can be performed with an 80 percent level in the hot leg piping. This is about 9 inches above the actual hot leg piping midplane elevation. (The hot leg piping has a 31-inch inside diameter.)

- A specially designed vortex breaker is used for the normal residual heat removal system suction nozzle. This vortex breaker connects vertically to the bottom of the hot leg piping. The normal residual heat removal system suction piping is connected to the bottom of this vortex breaker. With the vortex breaker, the amount of air entrainment remains below 10 percent unless the hot leg is essentially drained. Therefore, the potential for a loss of normal residual heat removal system flow and damage to the normal residual heat removal pump is substantially reduced.

- The normal residual heat removal pump suction piping is designed to be self-venting by sloping the lines continuously upward from the pump to the hot leg connection at the vortex breaker. If the pump should stop during midloop operation, any air bubbles present in the pump or suction piping are vented back up through the suction line to the water surface in the hot leg. This feature allows the operator to rapidly restart the pump with an air-free suction line.

- The normal residual heat removal pumps are designed to minimize cavitation and other adverse conditions when operating with minimal subcooling of the reactor coolant. Specifically, the plant piping layout configuration (such as piping elevations and routing) and the available and required pump net positive suction head characteristics allow the normal residual heat removal pumps to be started and operated at their full design flow rates, with saturation conditions in the reactor coolant system (associated with boiling in the reactor vessel). Therefore, the normal residual heat removal system is readily restored after a temporary loss of decay heat removal.

- The core makeup tanks, accumulators, and the in-containment refueling water storage tank are isolated, but can be manually actuated during midloop operations. In addition, the in-containment refueling water storage tank is automatically actuated on a sustained loss of shutdown decay heat removal. This arrangement provides a reliable water source for maintaining the reactor coolant system inventory that is either automatically or manually actuated.

- Redundant narrow-range level instrumentation indicates the reactor coolant system water level between the bottom of the hot leg and the top of the steam generator inlet elbow. Indication and low level alarms are provided in the main control room. In addition, this instrumentation actuates the in-containment refueling water storage tank makeup.

- Wide-range pressurizer level instrumentation used during cold plant operations is expanded to the bottom of the hot legs. This provides a continuous level indication in the main control room, from the normal level in the pressurizer to the range of the two narrow-range hot leg level instrumentation.

- Normal residual heat removal system heat exchanger discharge flow instrumentation provides main control room indication of return flow to the reactor vessel. A low-flow alarm alerts the operator to a decrease in normal residual heat removal system return flow from either heat exchanger.

- The drain-down of the reactor coolant system to the midloop operating level and the subsequent reactor coolant system inventory control during midloop operation are performed by the operator from the main control room.

The plant design precludes the need to locally coordinate actions in the containment with the main control room operators to control the reactor coolant system drain-down rate and level.

- Reactor coolant system hot leg wide range temperature instruments are provided in each hot leg. The orientation of the wide range thermowell-mounted resistance temperature detectors enable measurement of the reactor coolant fluid in the hot leg when in reduced inventory conditions. In addition, at least two incore thermocouple channels are available to directly measure the core exit temperature during midloop residual heat removal operation. These two thermocouple channels are associated with separate electrical divisions.

- The automatic depressurization system first-, second-, and third-stage valves, connected to the top of the pressurizer, are open whenever the core makeup tanks are blocked during shutdown conditions while the reactor vessel upper internals are in place. This provides a vent flow path to preclude pressurization of the reactor coolant system during shutdown conditions when decay heat removal is lost. This also allows the in-containment refueling water storage tank to automatically provide injection flow if it is actuated on a sustained loss of decay heat removal.

Administrative controls require containment closure capability in modes 5 and 6, during reduced inventory operations, and when the upper internals are in place. Containment closure capability is defined as the capability to close the containment prior to core uncovery following a loss of the normal decay heat removal system (that is, normal residual heat removal system). The containment design also includes penetrations for temporary cables and hoses needed for shutdown operations. These penetrations are isolated in an emergency.

In addition to these design features, appropriate procedures are defined to guide and direct the operator in the proper conduct of midloop operation and to aid in identifying and correcting abnormal conditions that might occur during shutdown operations.

1.9.5.1.5 Station Blackout

NRC Position:

The NRC has issued NUREG-0649 (Reference 34), NUREG-1032 (Reference 35), and NUREG-1109 (Reference 36) to address the unresolved safety issue of station blackout (USI-44). See subsection 1.9.4 for a discussion of USI-44.

To resolve this issue, the NRC published 10 CFR 50.63 and Regulatory Guide 1.155, which establish new requirements so that an operating plant can safely shut down following a loss of all ac power. SECY-94-084 (Reference 67), discusses station blackout for passive plants.

AP1000 Response:

The AP1000 is in conformance with the NRC guidelines for station blackout.

The AP1000 design minimizes the potential risk contribution of station blackout by not requiring ac power sources for design basis events. Safety-related systems do not need nonsafety-related ac power sources to perform safety-related functions.

The AP1000 safety-related passive systems automatically establish and maintain safe shutdown conditions for the plant following design basis events, including an extended loss of ac power sources. The passive systems can maintain these safe shutdown conditions after design basis events, without operator action, following a loss of both onsite and offsite ac power sources. Subsection 1.9.5.4 provides additional information on long-term actions following an extended station blackout beyond 72 hours.

The AP1000 also includes redundant nonsafety-related onsite ac power sources (diesel-generators) to provide electrical power for the nonsafety-related active systems which provide defense in depth.

AP1000 design features that mitigate the consequences of a station blackout are as follows:

- A full-load rejection capability to reduce the probability of loss of onsite power
- Safety-related passive residual heat removal heat exchanger
- Safety-related passive containment cooling
- Bleed and feed capability, using the safety-related automatic depressurization system in conjunction with the water available from the core makeup tanks, the accumulators, and the in-containment refueling water storage tank
- Class 1E batteries sized for 72 hours of operation under station blackout conditions
- Nonsafety-related reserve auxiliary transformers to provide power to selected ac power systems
- A nonsafety-related ac power system that includes two diesel-generators that automatically start on loss of offsite power
- An automatic nonsafety-related load-sequencing circuit that starts the following redundant nonsafety-related equipment after a loss of offsite power, once the associated diesel-generator is started:
  - Startup feedwater pump
  - Component cooling water pump
  - Service water pump
- Reactor coolant pumps without shaft seals

- Passive cooling for the rooms containing equipment assumed to operate during station blackout conditions (the protection and safety monitoring system cabinet rooms and the main control room) so that this equipment continues to operate. (Section 6.4 provides additional information.)

1.9.5.1.6 Fire Protection

NRC Position:

Current fire protection criteria are contained in GDC 3 and 10 CFR 50.48, guidelines for compliance with these criteria are provided in the Standard Review Plan, Section 9.5.1, including Branch Technical Position CMEB 9.5-1. Reference 9 identifies the following enhancements:

- Alternative, dedicated shutdown capability for main control room fires.

- Safe shutdown capability required for a fire in any other fire area, without reliance on any equipment in that area or re-entry into that area for repairs or for performance of operator actions.

- Fire protection for redundant shutdown systems in the reactor containment building must be provided to ensure, to the extent practicable, that on shutdown the division will be free of fire damage.

- Migration of smoke, hot gases, or fire-suppressant chemicals into other applicable fire areas must be minimized by design to prevent any adverse impact on safe shutdown capability, including operator actions.

SECY-98-161 (Reference 66) presents the results of the NRC review of the AP1000 Fire Protection System.

AP1000 Response:

Enhanced fire protection has been one of the goals of the AP1000 design. The following physical separation philosophy is used:

Outside Containment:

- Within the nuclear island, redundant divisions of safety-related equipment outside containment are located in safety-related areas separated from each other and from other areas in the plant by fire barriers with a minimum fire resistance rating of 3-hours to provide that safe shutdown can be achieved. Since most safety-related mechanical equipment is located inside containment, this applies primarily to the protection and safety monitoring system and the Class 1E dc and UPS system.

- Each safety-related area is provided with ventilation isolation provisions at the fire barrier boundaries to minimize the migration of smoke, hot gasses, or fire suppressant chemicals into other safety-related areas. Fiber-optic cables are used to provide communication between redundant protection and safety monitoring divisions.

- Exceptions to the use of three-hour fire barriers outside containment are made only in cases where physical separation conflicts with other requirements or where the equipment is not clearly division oriented, such as the main control room, the remote shutdown room, the main steam tunnel, and the passive containment cooling system valve room.

Inside Containment:

- The containment is a single fire area. Separation by three-hour fire barriers inside containment is not practical due to issues of hydrogen venting, compartment pressure equalization, and during high-energy line breaks and for system functionality. To the extent practical, separation is provided between redundant safety-related equipment.

- Separation between redundant safety-related equipment is accomplished by using existing structural walls. Where this is not possible, other methods are used, such as physical separation with no intervening combustibles.

- To the extent practical, the containment is split into two different fire zones for the purpose of routing of protection and safety monitoring system cabling and electrical power cabling. Divisions A and C cabling is routed below the operating deck, while Divisions B and D cabling is routed above the operating deck. Additional separation is provided by existing floors and walls and by the physical separation of cabling runs. Protection for the primary input sensors and the final actuation devices is accomplished by the physical separation of the various sensors and components using existing containment walls as barriers.

- The in-containment fire area contains reduced combustible material due to the use of sealless reactor coolant pump motors that do not use oil lubrication and due to strict combustible material limitations.

Main Control Room:

- Functionality requirements dictate that the main control room be a single fire zone. Features are included in the main control room to:

  – Reduce the probability of fire initiation
  – Reduce the likelihood of fire spreading
  – Increase the probability of fire detection
  – Effectively mitigate the effects of a fire

- In the event of main control room evacuation, safe shutdown conditions are established and maintained using the remote shutdown workstation.

See Appendix 9A.3 for information on the main steam tunnel and the passive containment cooling system valve room. See subsection 9.5.1 and Appendix 9A for additional information.

1.9.5.1.7 Intersystem LOCA

NRC Position:

Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release.

It is the NRC position that designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue. The Staff Requirements Memorandum to SECY-90-016 (Reference 31) added that consideration should be given to all elements of the low-pressure system (such as instrument lines, pump seals, heat exchanger tubes, and valve bonnets). For interfacing systems not designed to withstand full reactor coolant system pressure, it is necessary to provide leak testing capability for the pressure isolation valves, main control room position indication for de-energized reactor coolant system isolation valves, and high pressure alarms to alert control room operators when increasing reactor coolant system pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

AP1000 Response:

The AP1000 has incorporated various design features to address intersystem loss-of-coolant accident challenges. These design features result in very low AP1000 core damage frequency for intersystem loss-of-coolant accidents compared with operating nuclear power plants. The design features are primarily associated with the normal residual heat removal system and are discussed in Section 3 of WCAP-15993 (Reference 56) as well as DCD subsection 5.4.7. WCAP-15993 was prepared to document the evaluation of the AP1000 for conformance to the intersystem loss-of-coolant accident regulatory criteria identified in various NRC documents. See that document for additional information on conformance to intersystem loss-of-coolant accident regulatory criteria.

1.9.5.1.8 Hydrogen Generation and Control

NRC Position:

It is the NRC position that the likelihood of early containment failure from hydrogen combustion should be reduced. Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, advanced light water reactors should be designed to:

- Accommodate hydrogen equivalent to 100 percent metal-water reaction of the fuel cladding
- Limit containment hydrogen concentration to no greater than 10 percent

Further, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits, and given uncertainties in present analytical capabilities, advanced light water reactors should provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment.

AP1000 Response:

The AP1000 design includes mechanisms for monitoring and controlling hydrogen inside the containment. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the zircaloy cladding.

Passive autocatalytic hydrogen recombiners control hydrogen concentration following design basis events. Nonsafety-related hydrogen igniters control rapid releases of hydrogen during and after postulated events with degraded core conditions or with core melt.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

The containment air filtration system provides a capability to purge the containment atmosphere.

See subsection 6.2.4 for additional information.

1.9.5.1.9 Core-Concrete Interaction - Ability to Cool Core Debris

NRC Position:

Containment integrity could be breached in the event of a severe accident in which the core melts through the reactor vessel, resulting in interaction between core debris and concrete, which can generate large quantities of hydrogen and other gases. It is the NRC position that sufficient reactor cavity floor space be provided to enhance debris spreading, and that a method for quenching debris in the reactor cavity be incorporated. The NRC staff has not formulated specific criteria for debris bed coolability and reviews each vendor's design to determine how they address the general criteria for debris spreading and quenching.

AP1000 Response:

The AP1000 design provides superior protection against core-concrete interaction by reliably depressurizing the reactor vessel and flooding the reactor cavity to cool the vessel and prevent debris from relocating from the vessel into the containment. Based on the DOE/ARSAP analysis of the thermal-hydraulics of in-vessel debris retention (see Section 19.39 and Appendix 19B as supported by Theofanous, T. G., et al., Reference 62) performed using the Risk Oriented Accident Analysis Methodology, the AP1000 has a large margin to reactor vessel failure in the depressurized, flooded cavity condition. This strategy eliminates the large uncertainties associated with ex-vessel debris relocation that could result in containment failure even while meeting the NRC criteria for debris coolability in the cavity.

In the event that cavity flooding fails, the floor area under the vessel provides debris spreading area to enhance the coolability of the debris. The AP1000 containment design drains the water from the reactor coolant system, core makeup tanks and accumulators to the reactor cavity to provide enough water to quench ex-vessel debris. The heat is ultimately removed from the containment via the passive containment cooling system, and the condensate is returned to the cavity to continue to provide cooling water to the debris bed.

1.9.5.1.10 High Pressure Core Melt Ejection

NRC Position:

Direct containment heating associated with the ejection of molten core debris, under high pressure, from the reactor vessel can result in a rapid addition of energy to the containment atmosphere. It is the NRC position that, pending completion of ongoing research, it is prudent to provide protection against this potential failure mode. This protection should include the following two aspects:

- Providing a rate of reactor coolant system depressurization to preclude molten core ejection and creep rupture of steam generator tubes

- Arranging the reactor cavity so that high-pressure core debris ejection resulting from reactor vessel failure does not impinge on the containment boundary

AP1000 Response:

The AP1000 design includes an automatic depressurization system that is redundant, diverse, independent of ac power sources, and automatically actuated. The automatic depressurization system can also be manually actuated. Any of the automatic depressurization system lines can sufficiently reduce the reactor coolant system pressure to help preclude direct containment heating. Subsection 5.4.6 and Section 6.3 provide additional information on the automatic depressurization system.

In addition, the reactor cavity region and lower containment of the AP1000 are designed to preclude transport of significant core debris to the upper containment in the unlikely event of a high pressure melt ejection scenario from the reactor vessel. This is a passive feature involving the geometric configuration of the reactor cavity lower containment. There is no direct pathway from the cavity to the upper compartment.

1.9.5.1.11 Containment Performance

NRC Position:

The NRC opinion is that because there are substantial uncertainties in core damage predictions, and because it is very important to maintain defense in depth, it is necessary that the containment boundary serve as a reliable barrier against fission product release for credible severe accident challenges. Hence, a containment performance criterion has been proposed by the NRC.

The objective of the containment performance criterion is to provide a leaktight barrier against radioactive releases for two distinct categories of severe accident challenges:

- Rapid energy release, hydrogen combustion, and initial release of stored reactor coolant system energy

- Slow energy release, including decay heat and noncondensible gas generation, due to core-concrete interaction

The NRC position is that the reactor containment boundary should serve as a reliable barrier against fission product release for credible severe accident challenges. A conditional containment failure probability of 0.1 should be used unless a deterministic containment performance goal can offer comparable protection.

An alternate deterministic criterion proposed in SECY-90-016 (Reference 29) states that "...The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage..."

This capability should, to the extent practical, be provided by the passive capability of the containment and any related passive design features. The NRC further believes that following this 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

AP1000 Response:

The AP1000 design includes several features to minimize the potential for large fission product releases in the event of a severe accident. These features are aimed at both the prevention and the mitigation of severe accident phenomena that can threaten containment integrity. An adequate margin to containment performance is maintained.

The AP1000 containment is continuously cooled by natural air circulation outside the steel shell. During accident conditions, water drains on the outside of the containment vessel to increase heat transfer. The containment design best-estimate performance analysis alone shows that the maximum containment pressure reached maintains the containment shell stresses below the ASME Code Service Level C stress intensity limits, using a factor of safety of 1.5 for buckling of the top head.

Additionally, the probability of containment bypass scenarios is reduced by improved containment isolation, by designing to protect against interfacing system LOCAs, thereby reducing the associated core melt frequency, and by reducing the steam generator tube rupture core melt frequency.

The interfacing system LOCA core melt frequency is reduced by the use of several features, including effective leak testing of the normal residual heat removal system motor-operated isolation valves. A third valve is provided to the normal residual heat removal system suction line. It is a motor-operated valve located outside containment. This prevents inadvertently aligning the reactor coolant system to the normal residual heat removal system. The normal residual heat removal system design pressure is 900 psig. Therefore the ultimate rupture strength of the system prevents it from failing when exposed to the normal reactor coolant system operating pressure (2250 psia). See the position on intersystem LOCA for additional information on the normal residual heat removal system design against overpressurization.

Steam generator tube rupture core melt frequency is reduced by incorporating multiple levels of defense that are both redundant and diverse. The first level of defense relies on the use of nonsafety-related active systems and operator action. The second level of defense uses safety-related passive systems and equipment, such as the core makeup tanks and passive residual heat removal heat exchangers, without the safety-related automatic depressurization of the reactor coolant system. The third level of defense uses the redundant and diverse safety-related automatic depressurization system valves to depressurize the reactor coolant system and initiate low-pressure passive injection. Any of these levels of defense can prevent core damage during a steam generator tube rupture event.

Finally, containment isolation capabilities are substantially improved by reducing the number of penetrations and the number of open paths. Most of the open containment penetration lines use fail-closed valves for automatic isolation.

1.9.5.1.12 ABWR Containment Vent Design

This issue is specific to BWRs and PWRs with ice condenser containments. Therefore this issue does not apply to the AP1000 design.

1.9.5.1.13 Equipment Survivability

NRC Position:

Safety-related equipment used to mitigate design basis events is subject to a comprehensive set of criteria such as redundancy, diversity, environmental qualification, and quality assurance to provide reasonable assurance that they perform their intended functions, if needed. However, equipment used to mitigate the effects of severe accidents should not be treated in the same manner because of large differences in the likelihood of occurrence. There should be reasonable assurance that the equipment will operate in the severe accident environment for which they are intended and over the time span for which they are needed. However, equipment provided only for severe accident protection need not be subject to the 10 CFR 50.49, environmental qualification requirements, 10 CFR 50, Appendix B quality assurance requirements, and 10 CFR 50 Appendix A, redundancy and diversity requirements.

AP1000 Response:

The equipment used to mitigate severe accidents is identified in the AP1000 PRA evaluation report. Because of the nature of the passive safety features of the AP1000, there is very little equipment in this category. Equipment used to mitigate severe accidents is designed to survive the environmental conditions identified in the AP1000 PRA evaluation.

1.9.5.1.14 Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

NRC Position:

Currently, 10 CFR 100 requires that the magnitude of the operating basis earthquake be at least one-half that of the safe shutdown earthquake. This forces the safety-related system design at some plants to be controlled by the operating basis earthquake, but the NRC agrees that the operating basis earthquake should not control the safety-related system design. Therefore, the NRC recommends eliminating the operating basis earthquake from the design of systems, structures, and components. Until final rulemaking is approved for 10 CFR 100, Appendix A, the elimination of the operating basis earthquake from the design of passive plants will require an exemption from current regulations, with acceptable supporting justification from the designer. The details of this process will be resolved with the NRC through the appropriate code-related activities or supplemental regulatory guidance.

AP1000 Response:

The operating basis earthquake is not used as a design basis for AP1000 safety-related structures, systems, and components. For safety-related equipment, the safe shutdown earthquake is used as the design basis. 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.

Cyclic stresses due to earthquakes are included in the design of those components sensitive to fatigue. Analysis methods and allowable stresses provide margin for the design requirements for the safe shutdown earthquake. Sections 3.7 and 3.10 provide additional information.

1.9.5.1.15 In-Service Testing of Pumps and Valves

NRC Position:

Periodic testing according to ASME Code, Section XI is required to confirm operability of safety-related pumps and valves. The NRC believes that these testing requirements do not necessarily verify the capability of the components to perform their intended safety function. To address this concern, the NRC has issued Generic Letters 89-04 (Reference 38) and 89-10 (Reference 39), and has proposed rulemaking to extend in-service testing beyond code components and to demonstrate capability to perform safety functions. Reference 29 includes the following provisions to be applied to safety-related pumps and valves (not limited to only ASME Code Class 1, 2, or 3):

- Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design basis differential pressure.
- Check valve testing should incorporate the use of advanced, nonintrusive techniques to address degradation and performance characteristics.

- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, nonintrusive techniques.

In June 1990, the NRC position was approved, additionally noting that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

The NRC concluded that this was an issue for passive plant designs in SECY-94-084 (Reference 67), because the safety-related passive systems rely on the proper operation of equipment such as check valves and depressurization valves to mitigate the effects of transients.

AP1000 Response:

The AP1000 safety-related passive systems include the following design features:

- The AP1000 does not include any safety-related pumps.

- The motor-operated valve design is simplified by extending opening and closing times and by using simplified, conservative valve designs.

- Safety-related motor-operated valves are designed to be cycled with the plant at power.

- Features are included in the design to provide proper operational testing of the appropriate check valves, motor-operated valves, and air-operated valves, including flow and differential pressure testing during shutdown conditions.

Subsection 3.9.8.4 defines the responsibility for the in-service testing program for ASME Code Class 1, 2, and 3 valves.

Subsection 3.9.6 summarizes the requirements for the in-service testing program, including industry standards and NRC recommendations. The AP1000 system and valve designs generally allow implementation of the NRC recommendations in Generic Letters 89-04 and 89-10. Requirements for nonsafety-related pumps and valves that support the operation of systems that preclude unnecessary operation of the safety-related passive systems are outlined in subsection 3.9.6.

The AP1000 in-service testing program provides for periodic testing of the safety-related passive system components. The safety-related passive system components and systems are designed to meet the intent of the ASME Code, Section XI, for in-service inspection.

The AP1000 is designed for the following basic types of in-service testing of safety-related components:

- Periodic functional testing of active components during power operation (such as cycling of specific valves)

- Periodic flow/differential pressure operability testing of active components

- Periodic leak testing of the containment isolation valves.
- Periodic system flow or heat transfer rate testing of passive safety-related injection or cooling features during plant shutdown

The passive system design includes specific features to support in-service test performance:

- Remotely operated valves can be exercised during plant operation.
- Level, pressure, flow, and valve position instrumentation is provided for monitoring passive system equipment during plant operation and testing.
- Permanently installed test lines and connections are provided for performance of the containment isolation valve leakage testing.

1.9.5.2 Other Evolutionary and Passive Design Issues

Other evolutionary and passive design issues were identified in Reference 30.

1.9.5.2.1 Industry Codes and Standards

NRC Position:

SECY-91-273 (Reference 40) discusses NRC concerns with the use of recently developed or modified design codes and industry standards that the ALWR vendors are using in applications, but that have not yet been reviewed by the NRC for acceptability. The NRC recommends using the newest codes and standards endorsed by the NRC in the review of passive design applications. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.

AP1000 Response:

When the AP1000 design is based on revisions of industry codes and standards later than those required by NRC regulation, such use is explicitly discussed in the appropriate DCD section. Use of codes and standards later than those recommended in NRC guidance documents is also discussed in the appropriate DCD section.

Appendix 1A discusses regulatory guide conformance. For those standards endorsed by regulatory guides and subsequently superseded by a more recent revision, when the later revision is used its use is discussed or indicated in Appendix 1A.

1.9.5.2.2 Electrical Distribution

NRC Position:

The Commission approved the recommendations in SECY-91-078 (Reference 41) for evolutionary plant designs to include the following:

1. An alternate power source for nonsafety-related loads unless design margins for loss of nonsafety-related loads are no more severe than turbine-trip-only events in current plants

2. At least one offsite circuit to each redundant safety division supplied directly from offsite power sources with no intervening nonsafety-related buses

The applicability of this issue to passive designs is discussed in SECY-94-084 (Reference 67).

AP1000 Response:

See the response to station blackout in subsection 1.9.5.1.

1.9.5.2.3 Seismic Hazard Curves and Design Parameters

NRC Position:

To assess the seismic risk associated with an ALWR design, EPRI proposed the use of generic bounding seismic hazard curves for sites in the central and eastern United States. EPRI proposes that these curves be used in the seismic PRA. NRC regulations do not require performance of a seismic PRA to determine site acceptability.

The NRC has compared the proposed EPRI ALWR seismic hazard bounding curve for rock sites to hazard curves derived by Lawrence Livermore National Laboratories (LLNL) using historical earthquake methodology in NUREG/CR-4885 and to hazard curves generated by EPRI for the Seabrook site. The LLNL hazard curves are generally higher than the EPRI results for the same sites.

The proposed EPRI bounding curve is exceeded for accelerations below 0.1g and the NRC questions the adequacy of the proposed EPRI bounding curve at higher peak accelerations. The NRC concludes that the EPRI bounding hazards curve is nonconservative and also that its use in a seismic PRA assessment would underpredict the core damage frequency. Therefore, the EPRI curves are not sufficiently conservative for ALWR designer use.

The Combined License applicant must demonstrate that site-specific seismic parameters meet the certified design parameters, or a site-specific analysis will be required to confirm site acceptability.

AP1000 Response:

The AP1000 includes a seismic margin assessment performed in lieu of a seismic PRA. The seismic margin assessment follows the guidelines established in NUREG-1407 (Reference 42). This assessment demonstrates that the AP1000, located at a site having the most severe seismic inputs meeting the AP1000 site interface requirements, has a seismic risk comparable to that at existing nuclear power plants.

1.9.5.2.4 Leak-Before-Break

NRC Position:

GDC 4 provides the basis for the leak-before-break (LBB) analysis that has been approved for PWR primary piping, and the pressurizer surge, accumulator, and residual heat removal piping. In addition, it has been used for primary piping inside containment and for piping at least 6 inches nominal diameter and for both austenitic and carbon steel (clad with stainless) materials.

The NRC will evaluate the acceptability in ALWR designs, based on the justification provided by a deterministic fracture mechanics analysis submitted as part of the design. The NRC concluded that the analyses should be based on specific data, such as piping geometry, materials, and piping loads. However, the analyses may incorporate an initial set of bounding values and preliminary stress analysis results during the design certification phase. Subsequent verification of the preliminary analysis will be required.

The LBB approach has established certain limitations for excluding piping susceptible to failure from degradation mechanisms. In addition, the LBB introduced acknowledged inconsistency in the design basis, but the NRC published clarifications for the intended treatment of the containment, emergency core cooling systems, and environmental qualification in the LBB application.

The NRC position on LBB for the AP1000 is presented in SECY-95-172 (Reference 68).

AP1000 Response:

The AP1000 incorporates the leak-before-break approach for most high-energy lines inside containment that are 6 inches in diameter or larger. Detailed methodology and criteria are defined in subsection 3.6.3 and are consistent with those accepted by the NRC on existing nuclear power plants.

1.9.5.2.5 Classification of Main Steam Line of Boiling Water Reactors (BWRs)

This issue is specific to BWRs and therefore does not apply to the AP1000 design.

1.9.5.2.6 Tornado Design Basis

NRC Position:

WASH-1300 (Reference 43) and Regulatory Guide 1.76 contain the current NRC regulatory position for design basis tornados. Based on a contractor review of Regulatory Guide 1.76, the NRC recommends a maximum tornado speed of 300 mph be used for design basis tornado for passive ALWR designs.

The tornado design basis requirements have been used in establishing structural requirements against effects not covered explicitly in review guidance such as Regulatory Guides or the SRP. The Combined License applicant will have to demonstrate that the design will also be sufficient to withstand other site hazards such as aviation crashes, nearby explosions, and explosion debris and missiles.

AP1000 Response:

The AP1000 is designed in accordance with the NRC recommendations for a maximum tornado wind speed of 300 mph, as described in Section 3.3. The AP1000 site interface defined in Chapter 2 provides information to evaluate other site hazards if appropriate.

1.9.5.2.7 Containment Bypass

NRC Position:

Reasonable efforts should be made to minimize the possibility of containment bypass leakage, and ALWR designs should allow for a certain amount of leakage in the containment design. The NRC is evaluating the need for containment spray for all ALWRs. The containment spray provides containment temperature and pressure suppression effects and scrubs the containment atmosphere of fission products, mitigating the effects on the fission product bypass distribution.

AP1000 Response:

Although the phenomenon described for this item is primarily applicable to BWRs, the AP1000 has a variety of design features that help to reduce the potential for containment bypass leakage.

The response to the containment performance issue in subsection 1.9.5 provides additional information pertaining to various improvements that help to reduce containment bypass.

The safety-related passive containment cooling system design also contributes to the containment performance. The system includes multiple flow paths to provide cooling water for containment during severe accident conditions. The containment is also capable of successfully removing core decay heat with air-cooling alone.

The containment has a significantly reduced number of penetrations. The number of normally open containment penetrations is also reduced. The result is a low containment leak rate and a low probability of bypass.

The response to intersystem LOCA in subsection 1.9.5.1 provides additional information pertaining to applicable AP1000 design features that reduce the potential for intersystem LOCA and the potential for containment bypass.

Improvements are made to the steam generator design, such as the use of improved tube materials and tube supports. These improvements reduce the potential for tube leakage, which contributes to a reduction in containment bypass. Subsection 5.4.2 provides additional information on the steam generator design.

During a steam generator tube rupture event, the safety-related passive core cooling system automatically mitigates the effects of the event, including automatic safety-related protection against steam generator overfill.

The safety-related passive core cooling system provides long-term pH control for the containment sump, which helps to reduce the levels of airborne radioactivity, thereby reducing the consequences of leakage from the containment. Section 6.3 includes additional information on the passive core cooling system.

The diverse actuation system includes containment isolation features to provide isolation for the most risk-significant containment penetrations. PRA Chapter 24 discusses the provisions for isolating risk significant containment penetrations.

The performance of the passive fission product removal process and minimal potential for containment bypass precludes the need for a safety-related containment spray system on AP1000.

1.9.5.2.8 Containment Leak Rate Testing

NRC Position:

SECY-91-348 (Reference 44) proposes changes to 10 CFR 50, Appendix J to allow an increased interval from 24 months to 30 months for Type C containment leakage rate tests, until rule change proceedings are completed.

AP1000 Response:

10 CFR 50 Appendix J has been revised since SECY-91-348 was issued. AP1000 type C testing and compliance with 10 CFR 50 Appendix J is discussed in Section 6.2.5.

1.9.5.2.9 Post-Accident Sampling System

NRC Position:

Regulatory Guide 1.97 and NUREG-0737 (Reference 45) provide guidance regarding the design of the post-accident sampling system. 10 CFR 50.34 required the capability to obtain and analyze samples from containment and the reactor coolant system that may contain TID-14844 source term radioactive materials, without exceeding specified radiation exposures. The analysis and quantification are required for certain specified radionuclides that are indicators of the degree of core damage, containment hydrogen, dissolved gases, chloride, and boron concentrations.

The NRC concluded that adequate capability for monitoring post-accident hydrogen is provided by the safety-grade containment hydrogen monitoring instrumentation.

The NRC requires sampling the reactor coolant system for dissolved hydrogen, chloride, and oxygen. The time for taking these samples can be extended to 24 hours after the accident.

The NRC requires sampling the reactor coolant system for boron and for activity measurements. The time for taking these samples can be extended to 8 hours after power operation for boron and 24 hours after power operation for activity measurements.

AP1000 Response:

The post-accident sampling system is a subsystem of the primary sampling system, described in subsection 9.3.3.

The primary sampling system is designed to conform to the guidelines of the model Safety Evaluation Report on eliminating post-accident sampling system requirements from technical specifications for operating plants (Federal Register Volume 65, Number 211, October 31, 2000). The primary sampling system conforms with the most recent NRC position.

1.9.5.2.10 Level of Detail

NRC Position:

The Staff Requirements Memorandum for SECY-90-377 (Reference 47) provided guidance on the level of detail to be provided for a design certification application under 10 CFR 52. The guidance was that the application should include the information traditionally provided in a final safety analysis report, less the site-specific and as-procured information. This information should be supplemented by design inspections, tests, analysis, and acceptance criteria for those areas where the NRC is unable to make a final safety decision because of not having the site-specific information or the as-procured information, or because the technology is evolving so rapidly that it would be inappropriate to lock in the design.

AP1000 Response:

The AP1000 submittals are consistent with the requirements of 10 CFR 52 and the position in Reference 47.

1.9.5.2.11 Prototyping

NRC Position:

10 CFR 52.47 requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident conditions. Further, the interdependent effects among the safety features of the design must be found acceptable by analysis, appropriate test programs, experience, or a combination thereof. SECY-91-057 (Reference 48) informed the Commission of the steps the NRC was taking to identify the research needs for the AP600. SECY-91-074 (Reference 49) outlined the process the NRC would use to determine the need for a prototype or other demonstration facility for advanced reactor designs. SECY-91-273 (Reference 40) presented to the Commission the staff's recommendations for reviewing, monitoring and approving the Westinghouse test program to support the AP600 design certification application. SECY-92-030 (Reference 50) presented the Commission with the NRC opinion that there was a need for a full-height, full-pressure integral systems test to support the issuance of a final design approval.

AP1000 Response:

The Westinghouse testing program to assess the analytical methodologies used for the AP1000 safety analysis is described in Section 1.5 and is in conformance with the NRC position.

1.9.5.2.12 Inspections, Test, Analyses, and Acceptance Criteria (ITAAC)

NRC Position:

10 CFR 52 requires that the design certification application include the proposed tests, inspections, analyses, and the associated acceptance criteria. For certified standard designs, these tests, inspections, and analyses must apply to those portions of the facility covered by the design certification.

The Staff Requirements Memorandum for SECY-91-178 (Reference 51) provided guidance regarding development of ITAAC for final design approval and design certification applications.

AP1000 Response:

The AP1000 design certification application includes ITAACs.

1.9.5.2.13 Reliability Assurance Program

NRC Position:

SECY-89-013 (Reference 52) requires a reliability assurance program for design certification. The program would ensure that the design reliability of safety significant systems, structures, and components is maintained over the life of a plant.

The NRC is working on the development of a detailed guidance document consisting of two levels. The vendor submittal is the first level, consisting of a top-level program that identifies the scope, conceptual framework, and essential elements of an effective program. The Combined License applicant fully develops and implements the program based on the plant-specific design information.

AP1000 Response:

Section 16.2 includes a description of the reliability assurance program. The program description identifies the scope, conceptual framework, and essential elements of the program. The reliability assurance program confirms that the performance of the safety-related systems, structures, and components is consistent with the assumptions made for the design basis analysis.

In addition, the reliability assurance program monitors the long-term performance of important nonsafety-related structures, systems, and components that provide defense-in-depth against unnecessary actuation of the passive safety-related systems.

1.9.5.2.14 Site-Specific Probabilistic Risk Assessments (PRAs)

NRC Position:

10 CFR 52.47 requires all applicants for standard design certification to provide a PRA with enveloping analyses for seismic events and tornadoes. The Combined License applicant is

responsible for the site-specific PRA information that addresses site-specific events such as river flooding, storm surge, tsunami, volcanism, and hurricanes.

AP1000 Response:

The AP1000 PRA submitted as a part of the design certification application is based on a site that bounds a large percentage of plant sites in the United States and is described in Chapter 2. APP-GW-GLR-101 (Reference 73) identifies the potential external events that may impact the AP1000 risk on a site-specific basis. This technical report considers a wide range of site-specific external events as long as a site can show that the external events listed in this report bound those applicable to the site. The report also discusses impact of site selection on PRA Level 3 requirements.

1.9.5.2.15 Severe Accident Mitigation Design Alternatives

NRC Position:

The National Environmental Policy Act (NEPA) requires that alternatives be investigated for actions that may significantly affect the quality of the human environment. The timing of the NEPA hearing is at the Early Site Permit or Combined License stage. One objective of the 10 CFR 52 design certification rulemaking is to preclude changes to a certified standard plant design. The U.S. Court of Appeals has required the NRC to include consideration of severe accident mitigation design alternatives (SAMDAs) as a part of their environmental impact review for operating license applications. If this same process is followed for a plant design that had been certified, it may be necessary to reopen issues that had been resolved in the design certification rulemaking. To avoid this situation, the NRC issued SECY-91-229 (Reference 53) which recommended that SAMDAs be specifically addressed during the design certification rulemaking.

AP1000 Response:

The severe accident mitigation design alternatives (SAMDA) evaluation for AP1000 is contained in Appendix 1B.

1.9.5.2.16 Generic Rulemaking Related to Design Certification

NRC Position:

SECY-91-262 (Reference 54) provides the NRC recommendations to proceed with design-specific rulemaking where appropriate for passive designs, as information becomes available from ongoing efforts on those issues, independent of the design review and certification processes. In SECY-93-087 the NRC staff concludes that the design of passive plants is not sufficiently developed to determine whether generic rulemaking should be initiated for passive plant designs.

Generic rulemaking activities for source terms during severe accidents are ongoing, and the results may be used during design certification of the passive plants, focusing on updating 10 CFR 100 siting criteria, and planning to incorporate the revised source criteria in 10 CFR 50.

AP1000 Response:

No response necessary. See subsection 1.9.5.1.1 for a discussion of the use of a physically based source term.

1.9.5.3 Passive Design Issues

Issues related to the passive design were outlined in Reference 30.

1.9.5.3.1 Regulatory Treatment of Non-Safety Systems

NRC Position:

The NRC believes that its review of passive designs requires not only a review of the passive safety-related systems, but also a review of the functional capability and availability of the active nonsafety-related systems to provide significant defense-in-depth and accident and core damage prevention capability. The NRC issued a commission policy paper SECY-94-084 (Reference 67), on the regulatory treatment of non-safety systems (RTNSS), that outlines the process for resolving the RTNSS issue. This process includes a combination of probabilistic and deterministic criteria to identify risk-significant nonsafety-related systems.

AP1000 Response:

The AP1000 nonsafety-related active systems are designed to provide reliable support for normal plant operations and to provide defense-in-depth to minimize unnecessary challenges to the safety-related passive systems. These active systems are designed for more probable component and system failures. The systems include reliable, proven equipment and component designs. These active systems are capable of being powered by the nonsafety-related diesel-generators. The systems have nonsafety-related automatic actuation and controls that are separate from those of the safety-related systems.

These systems are designed to provide highly reliable performance. The design standards and operability provisions for these systems are discussed in subsection 3.2.2.6. Availability controls were developed for nonsafety related structures, systems, and components found to the important via the RTNSS process. The availability controls for the AP1000 are documented in DCD Section 16.3 and are the same as those for the AP600.

1.9.5.3.2 Definition of Passive Failure

NRC Position:

The NRC considered redefining failure of check valves in passive safety systems, where the valve fails to provide the mechanical movement to complete its intended safety function, to that of an active failure, as defined in Appendix A to 10 CFR 50. The NRC was concerned, since safety-related check valves in passive designs operate under different conditions (low flow and pressure without pump pressure to open valves) than current generation reactors and evolutionary designs. The check valves have increased safety significance to the operation of the passive safety-related systems, and operating experience has shown that they have a lower reliability than originally anticipated. The Staff position is described in SECY-94-084 (Reference 67).

AP1000 Response:

AP1000 is designed to tolerate the single failure of a check valve to change position to perform a safety-related function. Valve redundancy is provided for the core makeup tank discharge check valves (to close), the in-containment refueling water storage tank gravity injection check valves (to open), the containment recirculation gravity injection check valves (to open), and containment isolation line check valves (to close). The redundancy in the design for each of these safety-related flow paths is sufficient to accommodate the single failure of a check valve to reposition as required to perform its safeguards function.

Section 6.3 provides additional information on the failures assumed for the passive core cooling system including exceptions to the single failure criteria.

1.9.5.3.3 SBWR Stability

This issue is applicable to BWRs only.

1.9.5.3.4 Safe Shutdown Requirements

NRC Position:

GDC 34 requires that a residual heat removal system be provided to remove residual heat from the reactor core so that specified, acceptable fuel design limits are not exceeded. Regulatory Guide 1.139 and Branch Technical Position 5-1 implement this requirement and set forth conditions to cold shutdown (200°F for a PWR) using only safety-related systems within 36 hours.

The NRC evaluated the alternate means of addressing GDC 34 using passive safety-related systems to achieve a safe shutdown condition of 420°F. Additionally, the NRC reviewed the acceptability of using active, nonsafety-related systems to take a plant to cold shutdown conditions. The results of this review are presented in SECY-94-084 (Reference 67).

AP1000 Response:

The AP1000 includes safety-related passive systems and equipment that are designed to automatically establish and indefinitely maintain safe shutdown conditions for the plant following design basis events.

Sections 6.3 and 7.4 provide additional information pertaining to safe shutdown, using the safety-related passive systems.

1.9.5.3.5 Control Room Habitability

NRC Position:

10 CFR 50, Appendix A, GDC 19 requires adequate radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of five rem whole body, or its equivalent, to any part of the body, for the duration of the accident. Section 6.4 of the Standard Review Plan defines this dose criterion in terms of specific whole-body and organ doses (5 rem to whole body, and 30 rem each to thyroid and skin). The NRC requires that the analyses of main control room habitability be based on the dose criterion defined in GDC 19 of Appendix A to 10 CFR 50 and Section 6.4 of the Standard Review Plan (5 rem to whole body, and 30 rem each to thyroid and skin). In addition, the analyses of control room habitability should be based on the duration of the accident according to GDC 19 of Appendix A to 10 CFR 50.

AP1000 Response:

The AP1000 design includes a passive, safety-related main control room habitability system to meet the requirements of GDC 19. Section 6.4 provides additional information.

As described in subsection 15.6.5.3, the main control room operator doses following a design basis loss of coolant accident are within the dose criterion of GDC 19 (5 rem TEDE as applied to the AP1000 design).

1.9.5.3.6 Radionuclide Attenuation

NRC Position:

The NRC is concerned that use of the auxiliary building for holdup may require additional restrictions to be placed on the auxiliary building during normal operation. In addition, the NRC is continuing its evaluation of the need for a containment spray system for passive plant designs.

AP1000 Response:

The AP1000 design does not have a safety-related containment spray or take credit for auxiliary building holdup for mitigation of the design basis loss of coolant accident. The design includes a low-leakage-rate containment (0.10 percent per day) together with credit for aerosol removal by naturally occurring processes and pool scrubbing in containment. The low-leakage containment and natural aerosol removal are adequate to meet 10 CFR 50.34 dose limits, consistent with the physically-based source term.

1.9.5.3.7 Simplification of Off-Site Emergency Planning

NRC Position:

The NRC states that changes to emergency planning regulatory requirements may be appropriate, but that an NRC determination on this issue will require detailed design evaluation. Summaries of specific NRC conclusions are as follows:

- Unique characteristics of the designs should be considered in determining the extent of emergency planning, including the ability to prevent significant release of radioactive material or to provide delay times for all but the most unlikely events.
- A very low likelihood of all containment bypass sequences will be required before relaxing emergency planning requirements.
- Lack of information on source term and risk precludes further NRC evaluation of emergency preparedness for the passive plants at this time.
- Emergency planning requirements following the TMI-2 accident were not premised on specific assumptions regarding severe accident probability. So, as a policy matter, even very low calculated probabilities may not be a sufficient basis for changes to emergency planning requirements.

The industry and the NRC are working to determine a process, including developing technical criteria and methods, that would justify simplification of offsite emergency planning. The results of this process would be used as input to a generic rulemaking proposal to be initiated by nuclear industry organizations.

AP1000 Response:

The AP1000 PRA evaluation risk assessment includes calculations of the AP1000 response to severe accidents. This response includes the release of radionuclides following a severe accident. This analysis supports the technical basis for simplification of offsite emergency planning. The offsite emergency planning is discussed in Section 13.3.

1.9.5.4 Additional Licensing Issue

Post-72 Hour Support Actions

The AP1000 includes safety-related passive systems and equipment that are sufficient to automatically establish and maintain safe shutdown conditions for the plant following design basis events, assuming that the most limiting single failure occurs. The safety-related passive systems maintain safe shutdown conditions after an event -- without operator action, without onsite and offsite ac power sources.

The AP1000 includes nonsafety-related active systems and equipment designed to provide multiple levels of defense for a wide range of events. For the more probable events, these nonsafety-related systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related passive systems. These nonsafety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the standby nonsafety-related ac power sources is available.

Although event scenarios that result in an extended loss of the nonsafety-related systems or both offsite and onsite ac power sources for more than 72 hours are very unlikely, this potential is considered in the AP1000 design.

The actions described below are required following an extended loss of these nonsafety-related systems.

The safety functions required include the following:

- Core cooling, inventory, and reactivity control
- Containment cooling and ultimate heat sink
- Main control room habitability and post-accident monitoring
- Spent fuel pool cooling

The AP1000 design includes both onsite equipment and safety-related connections for use with transportable equipment and supplies to provide the following extended support actions:

- Provide electrical power to supply the post-accident and spent fuel pool monitoring instrumentation, using the ancillary diesel generators or a portable, engine-driven ac generator that both connect to electrical connections at the ancillary diesel generator electric panel. See Section 8.3 for additional information.

- Provide makeup water to the passive containment cooling water storage tank to maintain external containment cooling water flow, using one of the two PCS recirculation pumps powered by an ancillary diesel generator or a portable, engine-driven pump that connects to a safety-related makeup connection. See subsection 6.2.2 for additional information.

- Ventilation and cooling of the main control room, the instrumentation and control rooms, and the dc equipment rooms is provided by open doors and ancillary fans or portable fans powered by an ancillary diesel generator or a portable, engine-driven ac generator.

- Provide makeup water to the spent fuel pool from the passive containment cooling water storage tank, passive containment cooling water ancillary water storage tank, and from the long term makeup connection. See subsection 6.2.2.2.4 for a discussion of the operation of the passive containment cooling system and subsection 9.1.3.4.3 and 9.1.3.5 for discussion of makeup to the spent fuel pool.

- Provide a vent path between the fuel handling area and outside environment to vent water vapor generated by elevated spent fuel pool water temperature. See subsection 9.1.3.4.3.4 for additional information.

These actions are accomplished by the site support personnel, in coordination with the main control room operators. These actions are performed separate from, but in parallel with, other actions taken by the plant operators to directly mitigate the consequences of an event.

1.9.5.5 Operational Experience

Operational experience highlighted in NRC bulletins, generic letters, and information notices has been incorporated into the AP1000 design. Generic letters and bulletins are identified in WCAP-15800 (Reference 65). The applicability of each generic letter and bulletin to the AP1000 is assessed in WCAP-15800. If required, additional information for applicable issues is provided in the referenced sections of the DCD.

1.9.6 References

1. NUREG-0696, "Functional Criteria for Emergency Response Facilities," 1981.

2. Report NP-2770-LD, "EPRI PWR Safety Valve Test Report," December 1982.

3. NUREG-0933, "A Prioritization of Generic Safety Issues," June 2001.

4. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

5. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1984.

6. NRC letter to all PWR Licensees of Operating Reactors, Applicants for Operating Licensees and Holders of Construction Permits, and Ft. St. Vrain, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity," (Generic Letter 85-02) April 17, 1985.

7. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, September 1988.

8. NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," U.S. Nuclear Regulatory Commission, October 1983.

9. IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

10. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1980.

11. NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1979.

12. NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plant Stations," U.S. Nuclear Regulatory Commission, February 1981.

13. IEEE 344-1987, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

14. NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants."

15. NUREG-0660, "NRC Action Plan Developed as a result of the TMI-2 accident," May 1980.

16. NUREG-0985, "Nuclear Regulatory Commission Human Factors Program Plan Revision 2," April 1986.

17. IEEE 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers.

18. NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)" and NUREG-0933, "A Prioritization of Generic Safety Issues," June 1978.

19. NUREG-0484, Revision 1, "Methodology for Combining Dynamic Responses," May 1980.

20. IEEE 317-1983, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

21. ANSI/ANS-58.8-1984. "Time Response Design Criteria for Nuclear Safety Related Operator Actions."

22. NUREG/CR-2425, "Sediment and Radionuclide Transport in Rivers," December 1982.

23. NUREG-0693, "Analysis of Ultimate-Heat-Sink Cooling Ponds," November 1980.

24. NUREG-0733, "Analysis of Ultimate Heat-Sink Spray Ponds," August 1981.

25. NUREG-0858, "Comparison Between Field Data and Ultimate Heat Sink Cooling-Pond and Spray-Pond Models," September 1980.

26. ANSI 5.1, "Decay Heat Power in Light Water Reactors," American National Standards Institute, 1979.

27. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," September 1980.

28. ANSI 56.5-1979, "PWR and BWR Containment Spray System Design Criteria."

29. USNRC, SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues And Their Relationship to Current Regulatory Requirements," January 12, 1990.

30. NRC letter, Subject: Identification of Issues Concerning the Evolutionary and Passive Plant Designs, Dennis M. Crutchfield, USNRC Director, Division of Advanced Reactors and Special Projects, to E. E. Kintner, Chairman ALWR Steering Committee, February 27, 1992.

31. Staff Requirements Memorandum, Subject: SECY-90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, Samuel J. Chilk, USNRC Secretary, to James M. Taylor, USNRC Executive Director for Operations, June 26, 1990.

32. NUREG-1150, "Severe Accident Risk: An Assessment for Five U.S. Nuclear Power Plants," June 1989.

33. "Passive ALWR Source Term," D. E. Leaver, et al., DOE/ID-10321, February 1991.

34. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," Revision 1, September 1984.

35. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44," June 1988.

36. NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," June 1988.

37. Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," July 1986.

38. Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," April 3, 1989.

39. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," June 28, 1989.

40. SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," August 27, 1991.

41. SECY-91-078, "Chapter 11 of EPRI's Requirements Document and Additional Evolutionary Light Water Reactor Certification Issues," March 25, 1991.

42. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

43. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," October 1975.

44. SECY-91-348, preliminary untitled SECY related to containment leakrate testing, issued to the Commission for review, and not yet released by the NRC.

45. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

46. NUREG-4330, "Review of Light Water Reactor Regulatory Requirements, Volume 1, Identification of Regulatory Requirements That May Have Marginal Importance to Risk," April 1986.

47. SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," November 8, 1990.

48. SECY-91-057, "Early Review of AP600 and SBWR Research Needs," March 1, 1991.

49. SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," March 19, 1991.

50. SECY-92-030, "Integral System Testing Requirements for Westinghouse's AP600 Plant," January 27, 1992.

51. SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991.

52. SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," January 19, 1989.

53. SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," July 31, 1991.

54. SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," August 16, 1991.

55. Not used.

56. WCAP-15993, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," Revision 1, March 2003.

57. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," June 30, 1995.

58. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," L. Soffer, et al., February 1995.

59. SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," December 19, 1994.

60. SECY-94-194, "Proposed Revisions to 10 CFR Part 100 and 10 CFR Part 50, and New Appendix S to 10 CFR Part 50," July 27, 1994.

61. Not used.

62. Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

63. WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," Revision 1, August 2003.

64. NCRP Report No. 116, Limitation of Exposure to Ionizing Radiation, March 31, 1993.

65. WCAP-15800, "Operational Assessment for AP1000," Revision 3, July 2004.

66. SECY-98-161, "The Westinghouse AP1000 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems," July 1, 1998.

67. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.

68. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP1000 Standardized Passive Reactor Design," June 30, 1995.

69. WCAP-15992, "AP1000 Adverse Systems Interactions Evaluation Report," Revision 1, February 2003.

70. WCAP-15776, "Safety Criteria for the AP1000 Instrumentation and Control Systems," April 2002.

71. APP-GW-GLR-079, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Westinghouse Electric Company LLC.

72. APP-GW-GLR-040, "Plant Operations, Surveillance, and Maintenance Procedures," Westinghouse Electric Company LLC.

73. APP-GW-GLR-101, "AP1000 Probabilistic Risk Assessment External Events Evaluation to Support COL Applications," Westinghouse Electric Company LLC.

Table 1.9-1 (Sheet 1 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		**DCD Chapter, Section or Subsection**
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 2, 1970)	This regulatory guide is not applicable to AP1000.
1.2	Withdrawn	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-coolant Accident for Boiling Water Reactors (Rev. 2, June 1974)	This regulatory guide is not applicable to AP1000.
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors (Rev. 2, June 1974)	The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" will be followed instead of Reg. Guide 1.4.
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Rev. 0, March 10, 1971)	This regulatory guide is not applicable to AP1000.
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 10, 1971)	8.1 8.3.1 8.3.2 16.1 Bases
1.7	Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident (Rev. 2, November 1978)	6.1.1 6.2.4 15.6.3 Appendix 15A
1.8	Qualification and Training of Personnel for Nuclear Power Plants (Rev. 3, 1 May 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.9	Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (Proposed Rev. 3, November 1988)	This regulatory guide is not applicable to AP1000.
1.10	Withdrawn	
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 0, March 10, 1971)	3.6.2 6.2.3
1.12	Instrumentation for Earthquakes (Rev. 2, March 1997)	3.7.4

Table 1.9-1 (Sheet 2 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.13	Spent Fuel Storage Facility Design Basis (Proposed Rev. 2, December 1981)	9.1.2 9.1.3 9.1.4 16.1 Bases
1.14	Reactor Coolant Pump Flywheel Integrity (Rev. 1, August 1975)	5.4.1
1.15	Withdrawn	
1.16	Reporting of Operating Information - Appendix A Technical Specifications (Rev. 4, August 1975).	This regulatory guide is not applicable to AP1000 design certification.
1.17	Withdrawn	
1.18	Withdrawn	
1.19	Withdrawn	
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 2, May 1976)	3.9.2 14
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	11.5
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 17, 1972)	7.1 7.2 7.4
1.23	Onsite Meteorological Program (Second Proposed Rev. 1, April 1986)	2.3
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 23, 1972)	This regulatory guide is not applicable to AP1000.
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0, March 23, 1972)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.25.
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 3, February 1976)	3.2.2
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	6.2.2

Table 1.9-1 (Sheet 3 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	2.5 17
1.29	Seismic Design Classification (Rev. 3, September 1978)	3.2.1
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 11, 1972)	This regulatory guide is not applicable to AP1000 design certification.
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)	4.5.1 4.5.2 5.2.3 5.3.2 6.1.1
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Rev. 2, February 1977)	8.1 8.2 8.3.1 8.3.2 16.1 Bases
1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	3.11.2.1 3D.4.1.2 3D.6.4
1.34	Control of Electroslag Weld Properties (Rev. 0, December 28, 1972)	4.5.2 5.2.3 5.3.2
1.35	Inservice Inspection of Ungrouted Tendons in Pre-stressed Concrete Containments (Rev. 3, July 1990)	This regulatory guide is not applicable to AP1000.
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (Rev. 0, July 1990)	This regulatory guide is not applicable to AP1000.
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 23, 1973)	5.2.3 6.1.1
1.37	Quality Assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	17
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	17
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	17

Table 1.9-1 (Sheet 4 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 16, 1973)	This regulatory guide is not applicable to AP1000.
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments (Rev. 0, March 16, 1973)	14
1.42	Withdrawn	
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	5.2.3 5.3.2
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	4.5.1 4.5.2 5.2.3 5.3.2 6.1.1 10.3
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)	5.2.5 16.1 Bases
1.46	Withdrawn	
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	6.3 7.2 7.3 7.4 7.5 8.3.2
1.48	Withdrawn	
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	16
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	5.2.3 5.3.2 6.1.1
1.51	Withdrawn	
1.52	Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	6.4

Table 1.9-1 (Sheet 5 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)	7.1 7.2 7.4 15.2 15.3 15.4 15.5 15.6
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	6.1.2
1.55	Withdrawn	
1.56	Maintenance of Water Purity in Boiling Water Reactors (Rev. 1, July 1978)	This regulatory guide is not applicable to AP1000.
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 0, June 1973)	3.8.2 3.8.3
1.58	Withdrawn	
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	2.4 3.4
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	2.5 3.7.1
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)	3.7.1 3.9.23.10 Appendix 3D
1.62	Manual Initiation of Protective Actions (Rev. 0, October 1973)	7.1 7.2
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Task EE 405-4) (Rev. 3, February 1987)	8.3.1 Appendix 3D
1.64	Withdrawn	
1.65	Materials and Inspections for Reactor Vessel Closure Studs (Rev. 0, October 1973)	5.3.2
1.66	Withdrawn	
1.67	Withdrawn	

Table 1.9-1 (Sheet 6 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 2, August 1978)	14 16.1 Bases
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants (Rev. 1, January 1977)	This regulatory guide is not applicable to AP1000.
1.68.2	Initial Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, July 1978)	14
1.68.3	Preoperational Testing of Instrument and Air Control Systems (Task RS 709-4) (Rev. 0, April 1982)	9.3.1 14
1.69	Concrete Radiation Shields for Nuclear Power Plants (Rev. 0, December 1973)	3.8.4 12.3
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Rev. 3, November 1978)	1.1
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 0, December 1973)	5.2.3.4.6
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	This regulatory guide is not applicable to AP1000.
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	3.11 Appendix 3D
1.74	Withdrawn	
1.75	Physical Independence of Electric Systems (Rev. 2, September 1978)	7.1 7.2 7.3 7.4 7.5 8.1 8.3.1 8.3.2 9.5.1
1.76	Design Basis Tornado for Nuclear Power Plants (Rev. 0, April 1974)	2.3 3.3

Table 1.9-1 (Sheet 7 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.77. 16.1 Bases
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	2.2 6.4 9.4.1 9.5.1 16.1 Bases
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	14
1.80	Withdrawn	
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plant (Rev. 1, January 1975)	This regulatory guide is not applicable to AP1000.
1.82	Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident (Task 203-4) (Rev. 2, May, 1996)	6.3
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	5.4.2
1.84	Design and Fabrication Code Case Acceptability ASME Section III Division 1 (Rev. 32, June 2003)	4.5.1 4.5.2 5.2.1 5.2.3 10.3
1.85	Withdrawn	
1.86	Termination of Operating Licenses for Nuclear Reactors (Rev. 0, June 1974)	This regulatory guide is not applicable to AP1000 design certification.
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Rev. 1, June 1975)	This regulatory guide is not applicable to AP1000.
1.88	Withdrawn	

Table 1.9-1 (Sheet 8 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		**DCD Chapter, Section or Subsection**
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Task EE 042-2) (Rev. 1, June 1984)	3.11 Appendix 3D
1.90	Inservice Inspection of Prestressed Concrete Containment Structures With Grouted Tendons (Rev. 1, August 1977)	This regulatory guide is not applicable to AP1000.
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites (Rev. 1, February 1978)	19.58
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976; Rev. 2, July 2006)	3.7 Appendix 3D
1.93	Availability of Electric Power Sources (Rev. 0, December 1974)	8.1 8.3 16.1 Bases
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976)	This regulatory guide is not applicable to AP1000 design certification.
1.95	Withdrawn	
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Rev. 1, June 1976)	This regulatory guide is not applicable to AP1000.
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (Rev. 3, May 1983)	7.5 18.8 16.1 Bases Appendix 3D
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (Rev. 0, March 1976)	This regulatory guide is not applicable to AP1000.
1.99	Radiation Embrittlement of Reactor Vessel Materials (Task ME 305-4) (Rev. 2, May 1988)	5.3.2 5.3.3 16.1 Bases
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants (Task EE 108-5) (Rev. 2, June 1988)	3.10 Appendix 3D
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 3, August 1992)	This regulatory guide is not applicable to AP1000 design certification.
1.102	Flood Protection for Nuclear Power Plants (Rev. 1, September 1976)	3.4

Table 1.9-1 (Sheet 9 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.103	Withdrawn	
1.104	Withdrawn	
1.105	Instrument Setpoints for Safety-Related Systems (Task 1C 010-5) (Rev. 3, December 1999)	7.1 16
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1, March 1977)	8.1
1.107	Qualifications for Cement Grouting Tendons for Prestressing Tendons in Containment Structures (Rev. 1, February 1977)	This regulatory guide is not applicable to AP1000.
1.108	Withdrawn	
1.109	Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50 Appendix I (Rev. 1, October 1977)	11.3.3
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Rev. 0, March 1976)	11.2 11.3
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors (Rev. 1, July 1977)	2.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors (Rev. 0-R, May 1977)	11.2.3 11.3.3
1.113	Estimating Aquatic Dispersion of Effluents From Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, April 1977)	This regulatory guide is not applicable to AP1000 design certification.
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Rev. 2, May 1989)	This regulatory guide is not applicable to AP1000 design certification.
1.115	Protection Against Low-Trajectory Turbine Missiles (Rev 1, July 1977)	3.5 3.8.4
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	This regulatory guide is not applicable to AP1000 design certification.
1.117	Tornado Design Classification (Rev. 1, April 1978)	3.5 9.1.2

Table 1.9-1 (Sheet 10 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 3, April 1995)	7.1 8.1 8.3
1.119	Withdrawn	
1.120	Fire Protection Guidelines for Nuclear Power Plants (Rev. 1, November 1977)	9.5.1
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	5.4.2 16.1 Bases
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)	3.7 Appendix 3D
1.123	Withdrawn	
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Rev. 1, January 1978)	3.9.3, 9.1.2.1, 9.1.1.1
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	2.4
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	4.2
1.127	Inspection of Water-Control Structures Associated With Nuclear Power Plants (Rev. 1, March 1978)	This regulatory guide is not applicable to AP1000.
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, October 1978)	8.3.2
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, February 1978)	16.1 Bases
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 1, October 1978)	3.9.3
1.131	Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	3.11 Appendix 3D
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 1, March 1979)	This regulatory guide is not applicable to AP1000 design certification.
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	4.4.6.4

Table 1.9-1 (Sheet 11 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.134	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses (Rev. 3, March 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	2.4
1.136	Material for Concrete Containments (Rev. 2, June 1981)	This regulatory guide is not applicable to AP1000.
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, October 1979)	9.5.4
1.138	Laboratory Investigation of Soils for Engineering Analysis and Design of Nuclear Power Plants (Rev. 0, April 1978)	This regulatory guide is not applicable to AP1000 design certification.
1.139	Guidance for Residual Heat Removal (Rev. 0, May 1978)	6.3 7.4
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, June 2001)	9.4.1 9.4.4 9.4.5 9.4.7 9.4.9 16.1 Bases
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	6.2.4
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (Rev. 1, October 1981)	3.8.3 3.8.4 3.8.5
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	3.8.4 10.4.8 11.2 11.3 11.4 11.5
1.144	Withdrawn	
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 1, November 1982)	This regulatory guide is not applicable to AP1000 design certification.
1.146	Withdrawn	

Table 1.9-1 (Sheet 12 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1 (Rev. 12, May 1999)	5.2.4.3 6.6.3
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (Rev. 0, March 1981)	3.10 5.4.8
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations (Rev. 2, April 1996)	This regulatory guide is not applicable to AP1000 design certification.
1.150	Ultrasonic Testing of Reactor Pressure Vessel Welds During Preservice and Inservice Examinations (Rev. 1, February 1983)	5.2.4 5.3.2 5.3.4
1.151	Instrument Sensing Lines (Task 1C 126-5) (Rev. 0, July 1983)	7.1 7.5 7.6 7.7
1.152	Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants (Task 1C 127-5) (Rev. 1, January 1996)	7.1 7.2 7.3 7.4 7.5 7.6
1.153	Criteria for Power, Instrumentation, and Control Portions of Safety Systems (Task 1C 609-5) (Rev. 1, June 1996)	7.1 7.2 7.3 7.4 7.5 7.6
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (Rev. 0, January 1987)	5.3
1.155	Station Blackout (Task SI 501-4) (Rev. 0, August 1988)	8.2 8.3.1
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (Task EE 404-4) (Rev. 0, November 1987)	3.10 3.11 Appendix 3D
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance (Task RS 701-4) (Rev. 0, May 1989)	6.3

Table 1.9-1 (Sheet 13 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		**DCD Chapter, Section or Subsection**
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (Task EE 006-5) (Rev. 0, February 1989)	3.10 3.11 Appendix 3D
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 0, August 1990)	This regulatory guide is not applicable to AP1000 design certification.
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Rev. 2, March 1997)	3.8.6.5 17.5.6 The COL applicant is responsible for assessing conformance to Regulatory Guide 1.160 of monitoring the effectiveness of maintenance.
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb (Rev. 0, June 1995)	This regulatory guide is not applicable to AP1000 design certification.
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels (Rev. 0, February 1996)	This regulatory guide is not applicable to AP1000 design certification.
1.163	Performance Based Containment Leak-Test Program (Rev. 0, September 1995)	6.2 16.1 Bases
1.165	Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion (Rev. 0, March 1997)	
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions (Rev. 0, March 1997)	This regulatory guide is not applicable to AP1000 design certification.
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (Rev. 0, March 1997)	This regulatory guide is not applicable to AP1000 design certification.
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7

Table 1.9-1 (Sheet 14 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.174	An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis (Rev. 0, July 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.175	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing (Rev. 0, July 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.176	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance (Rev. 0, August 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications (Rev. 0, August 1998)	16.1 Bases
1.178	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping (Rev. 0, September 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors (Rev. 0, January 1999)	This regulatory guide is not applicable to AP1000 design certification.
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Rev. 1, October 2003)	Appendix 3D
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (Rev. 0, September 1999)	This regulatory guide is not applicable to AP1000 design certification.
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (Rev. 0, May 2000)	16.1 Bases

Table 1.9-1 (Sheet 15 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.183	Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors (Rev. 0, July 2000)	2.3 Appendix 3D 4.2 6.5.1 15.4 15.6.3 15.7 16.1 Bases
1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.185	Standard Format and Content for Post-shutdown Decommissioning Activities Report (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.186	Guidance and Examples of Identifying 10 CFR 50.2 Design Bases (Rev. 0, December 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments (Rev. 0, November 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.189	Fire Protection for Operating Nuclear Power Plants (Rev. 0, April 2001)	This regulatory guide is not applicable to AP1000 design certification.
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Rev. 0, March 2001)	5.3.2.6.2.2
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	9.4.1 6.4.5
1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	3.8.3.5 3.8.4.5.1 3.8.5.5

Table 1.9-2 (Sheet 1 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
TMI Action Plan Items			
I.A.1.1	Shift Technical Advisor	f	
I.A.1.2	Shift Supervisor Administrative Duties	f	
I.A.1.3	Shift Manning	f	
I.A.1.4	Long-Term Upgrading	f	See DCD subsection 13.1.1
I.A.2.1(1)	Qualifications - Experience	f	
I.A.2.1(2)	Training	f	
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	f	
I.A.2.2	Training and Qualifications of Operations Personnel	c	
I.A.2.3	Administration of Training Programs	f	
I.A.2.4	NRR Participation in Inspector Training	d	
I.A.2.5	Plant Drills	c	
I.A.2.6(1)	Revise Regulatory Guide 1.8	f	
I.A.2.6(2)	Staff Review of NRR 80-117	c	
I.A.2.6(3)	Revise 10 CFR 55	e	
I.A.2.6(4)	Operator Workshops	c	
I.A.2.6(5)	Develop Inspection Procedures for Training Programs	c	
I.A.2.6(6)	Nuclear Power Fundamentals	a	
I.A.2.7	Accreditation of Training Institutions	c	
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	f	
I.A.3.2	Operator Licensing Program Changes	c	
I.A.3.3	Requirements for Operator Fitness	c	
I.A.3.4	Licensing of Additional Operations Personnel	c	
I.A.3.5	Establish Statement of Understanding with INPO and DOE	d	

Table 1.9-2 (Sheet 2 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.A.4.1(1)	Short-Term Study of Training Simulators	c	
I.A.4.1(2)	Interim Changes in Training Simulators	f	
I.A.4.2(1)	Research on Training Simulators	f	
I.A.4.2(2)	Upgrade Training Simulator Standards	f	
I.A.4.2(3)	Regulatory Guide on Training Simulators	f	See DCD subsection 1.9.3, item (2)(i)
I.A.4.2(4)	Review Simulators for Conformance to Criteria	f	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	d	
I.A.4.4	Feasibility Study of NRC Engineering Computer	d	
I.B.1.1(1)	Prepare Draft Criteria	c	
I.B.1.1(2)	Prepare Commission Paper	c	
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	c	
I.B.1.1(4)	Review Responses to Determine Acceptability	c	
I.B.1.1(5)	Review Implementation of the Upgrading Activities	c	
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	e	
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	e	
I.B.1.2(1)	Prepare Draft Criteria	c	
I.B.1.2(2)	Review Near-Term Operating License Facilities	c	
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	c	
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	d	
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	d	
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	d	
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	d	

Table 1.9-2 (Sheet 3 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	d	
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Ensure Proper Testing and Return to Service	d	
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	d	
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	d	
I.B.2.1(6)	Observe Routine Maintenance	d	
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	d	
I.B.2.2	Resident Inspector at Operating Reactors	d	
I.B.2.3	Regional Evaluations	d	
I.B.2.4	Overview of Licensee Performance	d	
I.C.1(1)	Small Break LOCAs	f	
I.C.1(2)	Inadequate Core Cooling	f	
I.C.1(3)	Transients and Accidents	f	
I.C.1(4)	Confirmatory Analyses of Selected Transients	c	
I.C.2	Shift and Relief Turnover Procedures	f	
I.C.3	Shift Supervisor Responsibilities	f	
I.C.4	Control Room Access	f	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	g	See DCD subsection 1.9.3, item (3)(i)
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	f	
I.C.7	NSSS Vendor Review of Procedures	f	
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	f	

Table 1.9-2 (Sheet 4 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.C.9	Long-Term Program Plan for Upgrading of Procedures	c	See DCD subsections 13.5.1 and 1.9.3, item (2)(ii)
I.D.1	Control Room Design Reviews	g	See DCD subsection 1.9.3, item (2)(iii)
I.D.2	Plant Safety Parameter Display Console	g	See DCD subsection 1.9.3, item (2)(iv)
I.D.3	Safety System Status Monitoring	c	See DCD subsection 1.9.3, item (2)(v)
I.D.4	Control Room Design Standard	c	
I.D.5(1)	Operator-Process Communication	c	
I.D.5(2)	Plant Status and Post-Accident Monitoring	g	See DCD subsection 1.9.4, item I.D.5(2)
I.D.5(3)	On-Line Reactor Surveillance System	c	See DCD subsection 1.9.4, item I.D.5(3)
I.D.5(4)	Process Monitoring Instrumentation	c	
I.D.5(5)	Disturbance Analysis Systems	d	
I.D.6	Technology Transfer Conference	d	
I.E.1	Office for Analysis and Evaluation of Operational Data	d	
I.E.2	Program Office Operational Data Evaluation	d	
I.E.3	Operational Safety Data Analysis	d	
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	d	
I.E.5	Nuclear Plant Reliability Data Systems	d	
I.E.6	Reporting Requirements	d	
I.E.7	Foreign Sources	d	
I.E.8	Human Error Rate Analysis	d	
I.F.1	Expand QA List	c, j	See DCD subsections 1.9.4.2.1, item I.F.1 and 1.9.3, item (3)(ii)

Table 1.9-2 (Sheet 5 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	a	See DCD subsection 17.5
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	g	See DCD subsection 1.9.3, item (3)(iii)
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	g	See DCD subsection 1.9.3, item (3)(iii)
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	a	See DCD subsection 17.5
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	a	See DCD subsection 17.5
I.F.2(6)	Increase the Size of Licensees' QA Staff	f	
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	a	See DCD subsection 17.5
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	a	See DCD subsection 17.5
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	f	
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	a	See DCD subsection 17.5
I.F.2(11)	Define Role of QA in Design and Analysis Activities	a	See DCD subsection 17.5
I.G.1	Training Requirements	f, j	See DCD subsection 1.9.4.2.1, item I.G.1
I.G.2	Scope of Test Program	f, j	See DCD subsection 1.9.4.2.1, item I.G.2
II.A.1	Siting Policy Reformulation	c	
II.A.2	Site Evaluation of Existing Facilities	e	
II.B.1	Reactor Coolant System Vents	g	See DCD subsection 1.9.3, item (2)(vi)
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	g	See DCD subsection 1.9.3, item (2)(vii)

Table 1.9-2 (Sheet 6 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.B.3	Post-Accident Sampling	g	See DCD subsection 1.9.3, item (2)(viii)
II.B.4	Training for Mitigating Core Damage	f	
II.B.5(1)	Behavior of Severely Damaged Fuel	d	
II.B.5(2)	Behavior of Core Melt	d	
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structures	d	
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	f	
II.B.7	Analysis of Hydrogen Control	e	
II.B.8	Rulemaking Proceedings on Degraded Core Accidents	g	See DCD subsection 1.9.3, items (1)(i), (1)(xii), (2)(ix), (3)(iv), and (3)(v)
II.C.1	Interim Reliability Evaluation Program	c	
II.C.2	Continuation of Interim Reliability Evaluation Program	c	
II.C.3	Systems Interaction	e	
II.C.4	Reliability Engineering	c	
II.D.1	Testing Requirements	g	See DCD subsection 1.9.3, item (2)(x)
II.D.2	Research on Relief and Safety Valve Test Requirements	a	
II.D.3	Relief and Safety Valve Position Indication	g	See DCD subsection 1.9.3, item (2)(xi)
II.E.1.1	Auxiliary Feedwater System Evaluation	g	See DCD subsection 1.9.3, item (1)(ii)
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	g	See DCD subsection 1.9.3, items (1)(ii) and (2)(xii)
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	d, j	See DCD subsection 1.9.4.2.1, item II.E.1.3

Table 1.9-2 (Sheet 7 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.E.2.1	Reliance on ECCS	e	
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	c	
II.E.2.3	Uncertainties in Performance Predictions	a	
II.E.3.1	Reliability of Power Supplies for Natural Circulation	g	See DCD subsection 1.9.3, item (2)(xiii)
II.E.3.2	Systems Reliability	e	
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	e	
II.E.3.4	Alternate Concepts Research	c	
II.E.3.5	Regulatory Guide	e	
II.E.4.1	Dedicated Penetrations	g	See DCD subsection 1.9.3, item (3)(vi)
II.E.4.2	Isolation Dependability	g	See DCD subsection 1.9.3, item (2)(xiv)
II.E.4.3	Integrity Check	c	
II.E.4.4	Purging	g	See DCD subsection 1.9.3, item (2)(xv)
II.E.5.1	Design Evaluation	b	
II.E.5.2	B&W Reactor Transient Response Task Force	b	
II.E.6.1	Test Adequacy Study	d, j	See DCD subsection 1.9.4.2.1, item II.E.6.1
II.F.1	Additional Accident Monitoring Instrumentation	g	See DCD subsection 1.9.3, item (2)(xvii)
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	g	See DCD subsection 1.9.3, item (2)(xviii)
II.F.3	Instruments for Monitoring Accident Conditions	g	See DCD subsection 1.9.3, item (2)(xix)
II.F.4	Study of Control and Protective Action Design Requirements	a	

Table 1.9-2 (Sheet 8 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	d	
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	g	See DCD subsection 1.9.3, item (2)(xx)
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	c	
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	b	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	e	
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	d	
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	d	
II.J.1.2	Modify Existing Vendor Inspection Program	d	
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	d	
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	d	
II.J.2.1	Reorient Construction Inspection Program	d	
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	d	
II.J.2.3	Assign Resident Inspectors to All Construction Sites	d	
II.J.3.1	Organization and Staffing to Oversee Design and Construction	f	See DCD subsection 1.9.3, item (3)(vii)
II.J.3.2	Issue Regulatory Guide	e	
II.J.4.1	Revise Deficiency Reporting Requirements	f	
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	f	
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	b	
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	f	

Table 1.9-2 (Sheet 9 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(4)	Review Operating Procedures and Training Instructions	f	
II.K.1(5)	Safety-Related Valve Position Description	f	
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	f	
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	b	
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	b	
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	f	
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	f, j	See DCD subsection 1.9.4.2.1, item II.K.1(10)
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	f	
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	f	
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	f, j	See DCD subsection 1.9.4.2.1, item II.K.1(13)
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	f	
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	f	
II.K.1(16)	Implement Procedures That Identify PZR PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	f	
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	f, j	See DCD subsection 1.9.4.2.1, item II.K.1(17)

Table 1.9-2 (Sheet 10 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	b	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	b	
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	b	
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	b	
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	g	See DCD subsection 1.9.3, item (2)(xxi)
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	b	
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	e, j	See DCD subsection 1.9.4.2.1, item II.K.1(24)
II.K.1(25)	Develop Operator Action Guidelines	e	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	f	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	e	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	e	
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	b	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	b	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	b	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	b	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	b	
II.K.2(6)	Reevaluate Analysis of Dual-Level Setpoint Control	b	

Table 1.9-2 (Sheet 11 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.2(7)	Reevaluate Transient of September 24, 1977	b	
II.K.2(8)	Continued Upgrading of AFW System	e	
II.K.2(9)	Analysis and Upgrading of Integrated Control System	e	
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	b	See DCD subsection 1.9.3, item (2)(xxiii)
II.K.2(11)	Operator Training and Drilling	b	
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	e	
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	b	
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	b	
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	b	
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	g	See DCD subsection 1.9.3, item (1)(iii)
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	b	
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	e	
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	b	
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA	b	
II.K.2(21)	LOFT L3-1 Predictions	b	
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	g	See DCD subsection 1.9.3, item (1)(iv)
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	g	See DCD subsection 1.9.3, item (1)(iv)
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	f	

Table 1.9-2 (Sheet 12 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	e	
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	f, j	See DCD subsection 1.9.4.2.1, item II.K.3(5)
II.K.3(6)	Instrumentation to Verify Natural Circulation	e	
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	b	
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	e	
II.K.3(9)	Proportional Integral Derivative Controller Modification	g	See DCD subsection 1.9.4, item II.K.3(9)
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	f	
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	f	
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	f	
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	b	
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	b	
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	b	
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	b	
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	b	
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	g	See DCD subsection 1.9.3, item (1)(vii)
II.K.3(19)	Interlock on Recirculation Pump Loops	b	
II.K.3(20)	Loss of Service Water for Big Rock Point	b	

Table 1.9-2 (Sheet 13 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	b	
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	b	
II.K.3(23)	Central Water Level Recording	e	
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	b	
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	g	See DCD subsection 1.9.3, item (1)(iii)
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	e	
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	b	
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	g	See DCD subsection 1.9.3, item (1)(x)
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	b	
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	f	
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	f	
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	e	
II.K.3(33)	Evaluate Elimination of PORV Function	e	
II.K.3(34)	Relap-4 Model Development	e	
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	e	
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	e	
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	e	

Table 1.9-2 (Sheet 14 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	e	
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	e	
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	e	
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	e	
II.K.3(42)	Submit Requested Information on the Effects of Non Condensible Gases	e	
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	e	
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	b	
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	b	
II.K.3(46)	Response to List of Concerns from ACRS Consultant	b	
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	e	
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	e	
II.K.3(49)	Review of Procedures (NRC)	e	
II.K.3(50)	Review of Procedures (NSSS Vendors)	e	
II.K.3(51)	Symptom-Based Emergency Procedures	e	
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	e	
II.K.3(53)	Two Operators in Control Room	e	
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	e	
II.K.3(55)	Operator Monitoring of Control Board	e	
II.K.3(56)	Simulator Training Requirements	e	
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	b	

Table 1.9-2 (Sheet 15 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	f	
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	f	
III.A.1.2	Upgrade Licensee Emergency Support Facilities	g	See DCD subsection 1.9.3, item (2)(xxv)
III.A.1.3(1)	Maintain Supplies of Thyroid-Blocking Agent - Workers	c	
III.A.1.3(2)	Maintain Supplies of Thyroid-Blocking Agent - Public	c	
III.A.2.1(1)	Publish Proposed Amendments to the Rules	d	
III.A.2.1(2)	Conduct Public Regional Meetings	d	
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	d	
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	d	
III.A.2.2	Development of Guidance and Criteria	d	
III.A.3.1(1)	Define NRC Role in Emergency Situations	c	
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	c	
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	c	
III.A.3.1(4)	Prepare Commission Paper	c	
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	c	
III.A.3.2	Improve Operations Centers	c	
III.A.3.3	Communications	d	See DCD subsection 9.5.2.5.2
III.A.3.4	Nuclear Data Link	c	
III.A.3.5	Training, Drills, and Tests	c	
III.A.3.6(1)	Interaction of NRC and Other Agencies - International	c	
III.A.3.6(2)	Federal	c	

Table 1.9-2 (Sheet 16 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.A.3.6(3)	State and Local	c	
III.B.1	Transfer of Responsibilities to FEMA	c	
III.B.2(1)	The Licensing Process	c	
III.B.2(2)	Federal Guidance	c	
III.C.1(1)	Review Publicly Available Documents	d	
III.C.1(2)	Recommend Publication of Additional Information	d	
III.C.1(3)	Program of Seminars for News Media Personnel	d	
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	d	
III.C.2(2)	Provide Training for Members of the Technical Staff	d	
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	g	See DCD subsection 1.9.3, item (2)(xxvi)
III.D.1.1(2)	Review Information on Provisions for Leak Detection	a	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	a	
III.D.1.2	Radioactive Gas Management	a	
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	a	
III.D.1.3(2)	Review and Revise SRP	a	
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	a	
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	c	
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	a	
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	a	
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	a	
III.D.2.1(3)	Revise Regulatory Guides	a	

Table 1.9-2 (Sheet 17 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	c	
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	e	
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	e	
III.D.2.2(4)	Revise SRP and Regulatory Guides	e	
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	c	
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	c	
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	c	
III.D.2.3(4)	Prepare a Summary Assessment	c	
III.D.2.4(1)	Study Feasibility of Environmental Monitors	c	
III.D.2.4(2)	Place 50 TLDs Around Each Site	d	
III.D.2.5	Offsite Dose Calculation Manual	c	
III.D.2.6	Independent Radiological Measurements	d	
III.D.3.1	Radiation Protection Plans	c	
III.D.3.2(1)	Amend 10 CFR 20	d	
III.D.3.2(2)	Issue a Regulatory Guide	d	
III.D.3.2(3)	Develop Standard Performance Criteria	d	
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	d	
III.D.3.3	In-plant Radiation Monitoring	g	See DCD subsection 1.9.3, item (2)(xxvii)
III.D.3.4	Control Room Habitability	g	See DCD subsection 1.9.3, item (2)(xxviii)
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	d	

Table 1.9-2 (Sheet 18 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.D.3.5(2)	Investigate Methods of Obtaining Employee Health Data by Nonlegislative Means	d	
III.D.3.5(3)	Revise 10 CFR 20	d	
IV.A.1	Seek Legislative Authority	d	
IV.A.2	Revise Enforcement Policy	d	
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	d	
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	c	
IV.D.1	NRC Staff Training	d	
IV.E.1	Expand Research on Quantification of Safety Decision-Making	d	
IV.E.2	Plan for Early Resolution of Safety Issues	d	
IV.E.3	Plan for Resolving Issues at the CP Stage	d	
IV. E.4	Resolve Generic Issues by Rulemaking	d	
IV.E.5	Assess Currently Operating Reactors	c	
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	c	
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	c	
IV.G.1	Develop a Public Agenda for Rulemaking	d	
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	d	
IV.G.3	Improve Rulemaking Procedures	d	
IV.G.4	Study Alternatives for Improved Rulemaking Process	d	
IV.H.1	NRC Participation in the Radiation Policy Council	d	
V.A.1	Develop NRC Policy Statement on Safety	d	
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	d	
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	d	

Table 1.9-2 (Sheet 19 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
V.C.2	Study Need for Additional Advisory Committees	d	
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	d	
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	d	
V.D.2	Study Construction-During-Adjudication Rules	d	
V.D.3	Reexamine Commission Role in Adjudication	d	
V.D.4	Study the Reform of the Licensing Process	d	
V.E.1	Study the Need for TMI-Related Legislation	d	
V.F.1	Study NRC Top Management Structure and Process	d	
V.F.2	Reexamine Organization and Functions of the NRC Offices	d	
V.F.3	Revise Delegations of Authority to Staff	d	
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	d	
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	d	
V.G.1	Achieve Single Location, Long-Term	d	
V.G.2	Achieve Single Location, Interim	d	
Task Action Plan Items			
A-1	Water Hammer (former USI)	g	See DCD subsection 1.9.4, item A-1
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	g	See DCD subsection 1.9.4, item A-2
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	g	See DCD subsection 1.9.4, item A-3
A-4	CE Steam Generator Tube Integrity (former USI)	b	
A-5	B&W Steam Generator Tube Integrity (former USI)	b	
A-6	Mark I Short-Term Program (former USI)	b	

Table 1.9-2 (Sheet 20 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-7	Mark I Long-Term Program (former USI)	b	
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program (former USI)	b	
A-9	ATWS (former USI)	g	See DCD subsection 1.9.4, item A-9
A-10	BWR Feedwater Nozzle Cracking (former USI)	b	
A-11	Reactor Vessel Materials Toughness (former USI)	g	See DCD subsection 1.9.4, item A-11
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	g	See DCD subsection 1.9.4, item A-12
A-13	Snubber Operability Assurance	g	See DCD subsection 1.9.4, item A-13
A-14	Flaw Detection	a	
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	c	
A-16	Steam Effects on BWR Core Spray Distribution	b	
A-17	Systems Interactions in Nuclear Power Plants (former USI)	c, j	See DCD subsection 1.9.4.2.2, item A-17
A-18	Pipe Rupture Design Criteria	a	
A-19	Digital Computer Protection System	d	
A-20	Impacts of the Coal Fuel Cycle	d	
A-21	Main Steam Line Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	a	
A-22	PWR Main Steam Line Break - Core, Reactor Vessel and Containment Building Response	a	
A-23	Containment Leak Testing	d	
A-24	Qualification of Class e Safety-Related Equipment (former USI)	g	See DCD subsection 1.9.4, item A-24
A-25	Non-Safety Loads on Class e Power Sources	g	See DCD subsection 1.9.4, item A-25

Table 1.9-2 (Sheet 21 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-26	Reactor Vessel Pressure Transient Protection (former USI)	g	See DCD subsection 1.9.4, item A-26
A-27	Reload Applications	d	
A-28	Increase in Spent Fuel Pool Storage Capacity	g	See DCD subsection 1.9.4, item A-28
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	c, j	See DCD subsection 1.9.4.2.2, item A-29
A-30	Adequacy of Safety-Related DC Power Supplies	e	
A-31	RHR Shutdown Requirements (former USI)	g	See DCD subsection 1.9.4, item A-31
A-32	Missile Effects	e	
A-33	NEPA Review of Accident Risks	i	
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	e	
A-35	Adequacy of Offsite Power Systems	g	See DCD subsection 1.9.4, item A-35
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	g	See DCD subsection 1.9.4, item A-36
A-37	Turbine Missiles	a	
A-38	Tornado Missiles	a	
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	b	See DCD subsection 1.9.4, item A-39
A-40	Seismic Design Criteria - Short Term Program (former USI)	g	See DCD subsection 1.9.4, item A-40
A-41	Long Term Seismic Program	c	
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	b	
A-43	Containment Emergency Sump Performance (former USI)	g	See DCD subsection 1.9.4, item A-43
A-44	Station Blackout (former USI)	g	See DCD subsection 1.9.4, item A-44

Table 1.9-2 (Sheet 22 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-45	Shutdown Decay Heat Removal Requirements (former USI)	c	
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	g	See DCD subsection 1.9.4, item A-46
A-47	Safety Implications of Control Systems (former USI)	g	See DCD subsection 1.9.4, item A-47
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	g	See DCD subsection 1.9.4, item A-48
A-49	Pressurized Thermal Shock (former USI)	g	See DCD subsection 1.9.4, item A-49
B-1	Environmental Technical Specifications	d	
B-2	Forecasting Electricity Demand	d	
B-3	Event Categorization	a	
B-4	ECCS Reliability	e	
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	c, j	See DCD subsection 1.9.4.2.2, item B-5
B-6	Loads, Load Combinations, Stress Limits	e	
B-7	Secondary Accident Consequence Modeling	a	
B-8	Locking Out of ECCS Power Operated Valves	a	
B-9	Electrical Cable Penetrations of Containment	c	
B-10	Behavior of BWR Mark III Containments	b	
B-11	Subcompartment Standard Problems	d	
B-12	Containment Cooling Requirements (Non-LOCA)	c	
B-13	Marviken Test Data Evaluation	d	
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	e	
B-15	CONTEMPT Computer Code Maintenance	a	
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	e	

Table 1.9-2 (Sheet 23 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-17	Criteria for Safety-Related Operator Actions	c	See DCD subsection 1.9.4, item B-17
B-18	Vortex Suppression Requirements for Containment Sumps	e	
B-19	Thermal-Hydraulic Stability	c	
B-20	Standard Problem Analysis	d	
B-21	Core Physics	a	
B-22	LWR Fuel	a	See DCD subsection 1.9.4, item B-22
B-23	LMFBR Fuel	a	
B-24	Seismic Qualification of Electrical and Mechanical Components	e	
B-25	Piping Benchmark Problems	d	
B-26	Structural Integrity of Containment Penetrations	c	
B-27	Implementation and Use of Subsection NF	d	
B-28	Radionuclide/Sediment Transport Program	d	
B-29	Effectiveness of Ultimate Heat Sinks	d	See DCD subsection 1.9.4, item B-29
B-30	Design Basis Floods and Probability	d	
B-31	Dam Failure Model	a	
B-32	Ice Effects on Safety-Related Water Supplies	e	See DCD subsection 1.9.4, item B-32
B-33	Dose Assessment Methodology	d	
B-34	Occupational Radiation Exposure Reduction	e	
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	d	

Table 1.9-2 (Sheet 24 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	g	See DCD subsection 1.9.4, item B-36
B-37	Chemical Discharges to Receiving Waters	d	
B-38	Reconnaissance Level Investigations	a	
B-39	Transmission Lines	a	
B-40	Effects of Power Plant Entrainment on Plankton	a	
B-41	Impacts on Fisheries	a	
B-42	Socioeconomic Environmental Impacts	d	
B-43	Value of Aerial Photographs for Site Evaluation	d	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	d	
B-45	Need for Power - Energy Conservation	e	
B-46	Cost of Alternatives in Environmental Design	a	
B-47	Inservice Inspection of Supports - Classes 1, 2, 3, and MC Components	a	
B-48	BWR CRD Mechanical Failure (Collet Housing)	b	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	d	
B-50	Post-Operating Basis Earthquake Inspections	a	
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	e	
B-52	Fuel Assembly Seismic and LOCA Responses	e	
B-53	Load Break Switch	g	See DCD subsection 1.9.4, item B-53
B-54	Ice Condenser Containments	c	
B-55	Improved Reliability of Target Rock Safety Relief Valves	b	
B-56	Diesel Reliability	g	See DCD subsection 1.9.4, item B-56

Table 1.9-2 (Sheet 25 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-57	Station Blackout	e	
B-58	Passive Mechanical Failures	c	
B-59	(N-1) Loop Operation in BWRs and PWRs	d	
B-60	Loose Parts Monitoring System	c	
B-61	Allowable ECCS Equipment Outage Periods	g	See DCD subsection 1.9.4, item B-61
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	a	
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	g	See DCD subsection 1.9.4, item B-63
B-64	Decommissioning of Reactors	f	
B-65	Iodine Spiking	a	
B-66	Control Room Infiltration Measurements	g	See DCD subsection 1.9.4, item B-66
B-67	Effluent and Process Monitoring Instrumentation	e	
B-68	Pump Overspeed During LOCA	a	
B-69	ECCS Leakage Ex-Containment	e	
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	c	
B-71	Incident Response	e	
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	d	
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	e	
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	g	See DCD subsection 1.9.4, item C-1
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	c	

Table 1.9-2 (Sheet 26 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
C-3	Insulation Usage Within Containment	e	
C-4	Statistical Methods for ECCS Analysis	d	See DCD subsection 1.9.4, item C-4
C-5	Decay Heat Update	d	See DCD subsection 1.9.4, item C-5
C-6	LOCA Heat Sources	d	See DCD subsection 1.9.4, item C-6
C-7	PWR System Piping	c	
C-8	Main Steam Line Leakage Control Systems	b	
C-9	RHR Heat Exchanger Tube Failures	a	
C-10	Effective Operation of Containment Sprays in a LOCA	g	See DCD subsection 1.9.4, item C-10
C-11	Assessment of Failure and Reliability of Pumps and Valves	c	
C-12	Primary System Vibration Assessment	c	
C-13	Non-Random Failures	e	
C-14	Storm Surge Model for Coastal Sites	a	
C-15	NUREG Report for Liquids Tank Failure Analysis	a	
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	a	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	g	See DCD subsection 1.9.4, item C-17
D-1	Advisability of a Seismic Scram	a	
D-2	Emergency Core Cooling System Capability for Future Plants	a	
D-3	Control Rod Drop Accident	c	
New Generic Issues			
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	a	
2.	Failure of Protective Devices on Essential Equipment	a	

Table 1.9-2 (Sheet 27 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
3.	Set Point Drift in Instrumentation	c	
4.	End-of-Life and Maintenance Criteria	c	
5.	Design Check and Audit of Balance-of-Plant Equipment	e	
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	c	
7.	Failures Due to Flow-Induced Vibrations	a	
8.	Inadvertent Actuation of Safety Injection in PWRs	e	
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	e	
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	a	
11.	Turbine Disc Cracking	e	
12.	BWR Jet Pump Integrity	b	
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	a	
14.	PWR Pipe Cracks	c, j	See DCD subsection 1.9.4.2.3, item 14
15.	Radiation Effects on Reactor Vessel Supports	c	See DCD subsection 1.9.4, item 15
16.	BWR Main Steam Isolation Valve Leakage Control Systems	e	
17.	Loss of Offsite Power Subsequent to LOCA	a	
18.	Steam Line Break with Consequential Small LOCA	e	
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	e	
20.	Effects of Electromagnetic Pulse on Nuclear Power Plants	c	
21.	Vibration Qualification of Equipment	a	
22.	Inadvertent Boron Dilution Events	c, j	See DCD subsection 1.9.4.2.3, item 22
23.	Reactor Coolant Pump Seal Failures	c	See DCD subsection 1.9.4, item 23

Table 1.9-2 (Sheet 28 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
24.	Automatic Emergency Core Cooling System Switch to Recirculation	a, j	See DCD subsection 1.9.4.2.3, item 24
25.	Automatic Air Header Dump on BWR Scram System	b	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	e	
27.	Manual vs. Automated Actions	e	
28.	Pressurized Thermal Shock	e	
29.	Bolting Degradation or Failure in Nuclear Power Plants	c	See DCD subsection 1.9.4, item 29
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	a	
31.	Natural Circulation Cooldown	e	
32.	Flow Blockage in Essential Equipment Caused by Corbicula	e	
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	e	
34.	RCS Leak	a	
35.	Degradation of Internal Appurtenances in LWRs	a	
36.	Loss of Service Water	c	
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	e	
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	a	
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	e	
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	b	
41.	BWR Scram Discharge Volume Systems	b	
42.	Combination Primary/Secondary System LOCA	e	

Table 1.9-2 (Sheet 29 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
43.	Reliability of Air Systems	f, j	See DCD subsection 1.9.4.2.3, item 43
44.	Failure of Saltwater Cooling System	e	
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	g	See DCD subsection 1.9.4, item 45
46.	Loss of 125 Volt DC Bus	e	
47.	Loss of Off-Site Power	c	
48.	LCO for Class e Vital Instrument Buses in Operating Reactors	e	
49.	Interlocks and LCOs for Redundant Class e Tie Breakers	e	
50.	Reactor Vessel Level Instrumentation in BWRs	c	
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	g	See DCD subsection 1.9.4, item 51
52.	SSW Flow Blockage by Blue Mussels	e	
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	a	
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	e	
55.	Failure of Class e Safety-Related Switchgear Circuit Breakers to Close on Demand	a	
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	e	
57.	Effects of Fire Protection System Actuation	c	See DCD subsection 1.9.4, item 57
58.	Inadvertent Containment Flooding	a	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	d	
60.	Lamellar Tearing of Reactor Systems Structural Supports	e	
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	c	

Table 1.9-2 (Sheet 30 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
62.	Reactor Systems Bolting Applications	e	
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	a	
64.	Identification of Protection System Instrument Sensing Lines	c	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	e	
66.	Steam Generator Requirements	c	
67.2.1	Integrity of Steam Generator Tube Sleeves	d	
67.3.1	Steam Generator Overfill	e	
67.3.2	Pressurized Thermal Shock	e	
67.3.3	Improved Accident Monitoring	e, j	See DCD subsection 1.9.4.2.3, item 67.3.3
67.3.4	Reactor Vessel Inventory Measurements	e	
67.4.1	RCP Trip	e	
67.4.2	Control Room Design Review	e	
67.4.3	Emergency Operating Procedures	e	
67.5.1	Reassessment of SGTR Design Basis	d	
67.5.2	Reevaluation of SGTR Design Basis	d	
67.5.3	Secondary System Isolation	a	
67.6.0	Organizational Responses	e	
67.7.0	Improved Eddy Current Tests	e	
67.8.0	Denting Criteria	e	
67.9.0	Reactor Coolant System Pressure Control	e	
67.10.0	Supplement Tube Inspections	d	
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pumps Steam Supply Line Rupture	e	

Table 1.9-2 (Sheet 31 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
69.	Make-up Nozzle Cracking in B&W Plants	c	
70.	PORV and Block Valve Reliability	g	See DCD subsection 1.9.3, item (1)(iv)
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	a	
72.	Control Rod Drive Guide Tube Support Pin Failures	a	
73.	Detached Thermal Sleeves	a, j	See DCD subsection 1.9.4.2.3, item 73
74.	Reactor Coolant Activity Limits for Operating Reactors	a	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	g, j	See DCD subsection 1.9.4.2.3, item 75
76.	Instrumentation and Control Power Interactions	a	
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	e	
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	c	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	c	See DCD subsection 1.9.4.2.3, item 79
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	a	
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	a	
82.	Beyond Design Basis Accidents in Spent Fuel Pools	c, j	See DCD subsection 1.9.4.2.3, item 82
83.	Control Room Habitability	c	See DCD subsection 1.9.4.2.3, item 83
84.	CE PORVs	c	
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	a	
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	b	

Table 1.9-2 (Sheet 32 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
87.	Failure of HPCI Steam Line Without Isolation	g	See DCD subsection 1.9.4, item 87
88.	Earthquakes and Emergency Planning	c	
89.	Stiff Pipe Clamps	h (Medium)	
90.	Technical Specifications for Anticipatory Trips	a	
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	c	
92.	Fuel Crumbling During LOCA	a	
93.	Steam Binding of Auxiliary Feedwater Pumps	g	See DCD subsection 1.9.4, item 93
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	g	See DCD subsection 1.9.4, item 94
95.	Loss of Effective Volume for Containment Recirculation Spray	c	
96.	RHR Suction Valve Testing	e	
97.	PWR Reactor Cavity Uncontrolled Exposures	e	
98.	CRD Accumulator Check Valve Leakage	a	
99.	RCS/RHR Suction Line Valve Interlock on PWRs	f	
100.	OTSG Level	b	
101.	BWR Water Level Redundancy	c	
102.	Human Error in Events Involving Wrong Unit or Wrong Train	c	
103.	Design for Probable Maximum Precipitation	g	See DCD subsection 1.9.4, item 103
104.	Reduction of Boron Dilution Requirements	a	
105.	Interfacing Systems LOCA at BWRs	c	See DCD subsection 1.9.4, item 105
106.	Piping and Use of Highly Combustible Gases in Vital Areas	c	See DCD subsection 1.9.4, item 106
107.	Main Transformer Failures	a	

Table 1.9-2 (Sheet 33 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
108.	BWR Suppression Pool Temperature Limits	a	
109.	Reactor Vessel Closure Failure	a	
110.	Equipment Protective Devices on Engineered Safety Features	a	
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	d	
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	d	
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	c	See DCD subsection 1.9.4, item 113
114.	Seismic-Induced Relay Chatter	e	
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	c	
116.	Accident Management	a	
117.	Allowable Time for Diverse Simultaneous Equipment Outages	a	
118.	Tendon Anchorage Failure	f	
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	d	
119.2	Piping Damping Values	a	
119.3	Decoupling the OBE from the SSE	d	
119.4	BWR Piping Materials	d	
119.5	Leak Detection Requirements	d	
120.	On-Line Testability of Protection Systems	c, j	See DCD subsection 1.9.4.2.3, item 120
121.	Hydrogen Control for Large, Dry PWR Containments	c	See DCD subsection 1.9.4, item 121
122.1.a	Failure of Isolation Valves in Closed Position	e	
122.1.b	Recovery of Auxiliary Feedwater	e	
122.1.c	Interruption of Auxiliary Feedwater Flow	e	

Table 1.9-2 (Sheet 34 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
122.2	Initiating Feed-and-Bleed	c	
122.3	Physical Security System Constraints	a	
123.	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	a	
124.	Auxiliary Feedwater System Reliability	g	See DCD subsection 1.9.4, item 124
125.I.1	Availability of the STA	a	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	e	
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	e	
125.I.2.c	Need for Additional Protection Against PORV Failure	a	
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	e	
125.I.3	SPDS Availability	c	
125.I.4	Plant-Specific Simulator	a	
125.I.5	Safety Systems Tested in All Conditions Required by Design Basis Analysis	a	
125.I.6	Valve Torque Limit and Bypass Switch Settings	a	
125.I.7.a	Recover Failed Equipment	a	
125.I.7.b	Realistic Hands-On Training	a	
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	a	
125.II.1.a	Two-Train AFW unavailability	a	
125.II.1.b	Review Existing AFW Systems for Single Failure	e	
125.II.1.c	NUREG-0737 Reliability Improvements	a	
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	a	
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	a	

Table 1.9-2 (Sheet 35 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	a	
125.II.4	Thermal Stress of OTSG Components	a	
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	a	
125.II.6	Reexamine PRA-Based Estimates of the Likelihood of a Severe Core Damage Accident Based on Loss of All Feedwater	a	
125.II.7	Reevaluate Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	c	
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	a	
125.II.9	Enhanced Feed-and-Bleed Capability	a	
125.II.10	Hierarchy of Impromptu Operator Actions	a	
125.II.11	Recovery of Main Feedwater as Alternative to AFW	a	
125.II.12	Adequacy of Training Regarding PORV Operation	a	
125.II.13	Operator Job Aids	a	
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	a	
126.	Reliability of PWR Main Steam Safety Valves	d	
127.	Testing and Maintenance of Manual Valves in Safety-Related Systems	a	
128.	Electrical Power Reliability	h (High)	See DCD subsection 1.9.4, item 128
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	a	
130.	Essential Service Water Pump Failures at Multiplant Sites	f	See DCD subsection 1.9.4, item 130
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System in Westinghouse Plants	e	
132.	RHR Pumps Inside Containment	a	

Table 1.9-2 (Sheet 36 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	d	
134.	Rule on Degree and Experience Requirements	c	
135.	Steam Generator and Steam Line Overfill	c	See DCD subsection 1.9.4, item 135
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	d	
137.	Refueling Cavity Seal Failure	a	
138.	Deinerting Upon Discovery of RCS Leakage	a	
139.	Thinning of Carbon Steel Piping in LWRs	d	
140.	Fission Product Removal Systems	a	
141.	LBLOCA With Consequential SGTR	a	
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	c	See DCD subsection 1.9.4, item 142
143.	Availability of Chilled Water Systems	c, j	See DCD subsection 1.9.4.2.3, item 143
144.	Scram Without a Turbine/Generator Trip	a	
145.	Actions to Reduce Common Cause Failures	c	
146.	Support Flexibility of Equipment and Components	d	
147.	Fire-Induced Alternate Shutdown Control Room Panel Interactions	d	
148.	Smoke Control and Manual Fire-Fighting Effectiveness	d	
149.	Adequacy of Fire Barriers	a	
150.	Overpressurization of Containment Penetrations	a	
151.	Reliability of Recirculation Pump Trip During an ATWS	c	
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	a	
153.	Loss of Essential Service Water in LWRs	c, j	See DCD subsection 1.9.4.2.3, item 153

Table 1.9-2 (Sheet 37 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
154.	Adequacy of Emergency and Essential Lighting	a	
155.1	More Realistic Source Term Assumptions	g	
155.2	Establish Licensing Requirements For Non-Operating Facilities	d	
155.3	Improve Design Requirements For Nuclear Facilities	a	
155.4	Improve Criticality Calculations	a	
155.5	More Realistic Severe Reactor Accident Scenario	a	
155.6	Improve Decontamination Regulations	a	
155.7	Improve Decommissioning Regulations	a	
156	Systematic Evaluation Program	f	
157	Containment Performance	c	
158	Performance Of Safety-Related Power-Operated Valves Under Design Basis Conditions	c	
159	Qualification Of Safety-Related Pumps While Running On Minimum Flow	a	
160	Spurious Actuations Of Instrumentation Upon Restoration Of Power	a	
161	Use Of Non-Safety-Related Power Supplies In Safety-Related Circuits	a	
162	Inadequate Technical Specifications For Shared Systems At Multiplant Sites When One Unit Is Shut Down	a	
163	Multiple Steam Generator Tube Leakage	h (Medium)	See DCD subsection 1.9.4.2.3, item 163
164	Neutron Fluence In Reactor Vessel	a	
165	Spring-Actuated Safety And Relief Valve Reliability	c	
166	Adequacy Of Fatigue Life Of Metal Components	c	
167	Hydrogen Storage Facility Separation	a	
168	Environmental Qualification Of Electrical Equipment	f	See DCD subsection 1.9.4.2.3, item 168

Table 1.9-2 (Sheet 38 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
169	BWR MSIV Common Mode Failure Due To Loss Of Accumulator Pressure	a	
170	Fuel Damage Criteria For High Burnup Fuel	c	
171	ESF Failure From LOOP Subsequent To A LOCA	c	
172	Multiple System Responses Program	e	
173.A	Spent Fuel Storage Pool Operating Facilities	c	
173.B	Spent Fuel Storage Pool Permanently Shutdown Facilities	c	
174	Fastener Gaging Practices	c	
175	Nuclear Power Plant Shift Staffing	c	
176	Loss Of Fill-Oil In Rosemount Transmitters	c	
177	Vehicle Intrusion At TMI	g	
178	Effect Of Hurricane Andrew On Turkey Point	d	
179	Core Performance	c	
180	Notice Of Enforcement Discretion	d	
181	Fire Protection	d	
182	General Electric Extended Power Uprate	b	
183	Cycle-Specific Parameter Limits In Technical Specifications	d	
184	Endangered Species	d	
185	Control of Recriticality following Small-Break LOCA in PWRs	h (High)	See DCD subsection 1.9.4.2.3, item 185
186	Potential Risk and Consequences of Heavy Load Drops	a	
187	The Potential impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants.	a	
188	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	a	
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Concentration during a Severe Accident	a	

Table 1.9-2 (Sheet 39 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
190	Fatigue Evaluation Of Metal Components For 60-Year Plant Life	c	
191	Assessment Of Debris Accumulation On PWR Sump Performance	h (High)	See DCD subsections 6.3.2.2.7 and 1.9.4.2.3, item 191
Human Factors Issues			
HF1.1	Shift Staffing	f	
HF1.2	Engineering Expertise on Shift	c	
HF1.3	Guidance on Limits and Conditions of Shift Work	c	
HF2.1	Evaluate Industry Training	d	
HF2.2	Evaluate INPO Accreditation	d	
HF2.3	Revise SRP Section 13.2	d	
HF3.1	Develop Job Knowledge Catalog	d	
HF3.2	Develop License Examination Handbook	d	
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	e	
HF3.4	Examination Requirements	e	
HF3.5	Develop Computerized Exam System	d	
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	c, i	
HF4.2	Procedures Generation Package Effectiveness Evaluation	d	
HF4.3	Criteria for Safety-Related Operator Actions	e	
HF4.4	Guidelines for Upgrading Other Procedures	c, j	See DCD subsection 1.9.4.2.4, item HF4.4
HF4.5	Application of Automation and Artificial Intelligence	e	
HF5.1	Local Control Stations	c	See DCD subsection 1.9.4, item HF5.1
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	g	See DCD subsection 1.9.4, item HF5.2

Table 1.9-2 (Sheet 40 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
HF5.3	Evaluation of Operational Aid Systems	e	
HF5.4	Computers and Computer Displays	e	
HF6.1	Develop Regulatory Position on Management and Organization	e	
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	e	
HF7.1	Human Error Data Acquisition	d	
HF7.2	Human Error Data Storage and Retrieval	d	
HF7.3	Reliability Evaluation Specialist Aids	d	
HF7.4	Safety Event Analysis Results Applications	d	
HF8	Maintenance and Surveillance Program	c	
Chernobyl Issues			
CH1.1A	Symptom-Based EOPs	d	
CH1.1B	Procedure Violations	d	
CH1.2A	Test, Change, and Experiment Review Guidelines	d	
CH1.2B	NRC Testing Requirements	d	
CH1.3A	Revise Regulatory Guide 1.47	d	
CH1.4A	Engineered Safety Feature Availability	d	
CH1.4B	Technical Specification Bases	d	
CH1.4C	Low Power and Shutdown	d	
CH1.5	Operating Staff Attitudes Toward Safety	d	
CH1.6A	Assessment of NRC Requirements on Management	d	
CH1.7A	Accident Management	d	
CH2.1A	Reactivity Transients	d	
CH2.2	Accidents at Low Power and at Zero Power	e	
CH2.3A	Control Room Habitability	e	
CH2.3B	Contamination Outside Control Room	d	

Table 1.9-2 (Sheet 41 of 41)

LISTING OF UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
CH2.3C	Smoke Control	d	
CH2.3D	Shared Shutdown Systems	d	
CH2.4A	Firefighting With Radiation Present	d	
CH3.1A	Containment Performance	d	
CH3.2A	Filtered Venting	d	
CH4.1	Size of the Emergency Planning Zones	a	
CH4.2	Medical Services	a	
CH4.3A	Ingestion Pathway Protective Measures	d	
CH4.4A	Decontamination	d	
CH4.4B	Relocation	d	
CH5.1A	Mechanical Dispersal in Fission Product Release	d	
CH5.1B	Stripping in Fission Product Release	d	
CH5.2A	Steam Explosions	d	
CH5.3	Combustible Gas	a	
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	a	
CH6.1B	Structural Graphite Experiments	d	
CH6.2	Assessment	d	

Notes:

a. Issue has been prioritized as **Low, Drop** or has not been prioritized.
b. Issue is not an AP1000 design issue. Issue is applicable to GE, B&W, or CE designs only.
c. Issue resolved with no new requirements.
d. Issue is not a design issue (Environmental, Licensing, or Regulatory Impact Issue; or covered in an existing NRC program).
e. Issue superseded by one or more issues.
f. Issue is not an AP1000 design certification issue. Issue is applicable to current operating plants or is programmatic in nature.
g. Issue is resolved by establishment of new regulatory requirements and/or guidance.
h. Issue is unresolved pending generic resolution (for example, prioritized as **High, Medium**, or possible resolution identified).
i. The AP600 DSER (Draft NUREG-01512) identified this item as not being required to be addressed by 10 CFR 52.47.
j. The AP600 DSER (Draft NUREG-01512) identified this item as required to be discussed.