20 GENERIC ISSUES

In this chapter, the staff discusses its evaluation of (1) the compliance of the Westinghouse AP1000 design with 10 CFR 52.47(a)(1)(iv) and 52.47(a)(1)(ii), and (2) the incorporation of operating experience into the AP1000 design. The applicant for a standard design certification is required by 10 CFR 52.47(a)(1)(iv) to propose resolutions of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) defined in NUREG-0933, "A Prioritization of Generic Safety Issues," that are (1) technically relevant to the design and (2) identified in the applicable supplement to NUREG-0933 that was current 6 months prior to the application. In addition, the applicant is required under 10 CFR 52.47(a)(1)(ii) to propose resolutions to the technically relevant portions of Three Mile Island (TMI) Action Plan items addressed in 10 CFR 50.34(f).

Because a large number of issues are relevant to the AP1000 design, the staff grouped its evaluations into the following sections, according to the issue type in Appendix B of NUREG-0933:

- Section 20.2 contains the task action plan items.

- Section 20.3 contains the new generic issues.

- Section 20.4 contains the TMI Action Plan items.

- Section 20.5 contains the human factors issues.

- Section 20.6 lists the 50.34(f) TMI Action Plan items relevant to the AP1000 design.

- Section 20.7 discusses the incorporation of operating experience into the AP1000 design through generic communications.

20.1 Overview of Staff Conclusion

20.1.1 Compliance With 10 CFR 52.47(a)(1)(iv)

As stated above, an application for design certification must include proposed resolutions of those USIs and medium- and high-priority GSIs identified in the NUREG-0933 supplement that was current 6 months prior to the application, and which are technically relevant to the design.

The applicant made its application for the AP1000 standardized plant design in accordance with the provisions of 10 CFR 52.45, in the Design Control Document (DCD) Tier 2. The staff reviewed Supplement 14 to NUREG-0933 to identify the list of issues contained in Appendix B of NUREG-0933, "Applicability of NUREG-0933 Issues to Operating and Future Plants," that

should be addressed to conform to Section 52.47(a)(1)(iv). In addition, the staff added nine other issues (A-17, A-29, B-5, 14, 22, 29, 43, 82, and II.K.3(5)) that were resolved without the issuance of new requirements, but for which the staff had recommended the development of specific guidance for future plants.

The issues that need to be resolved to comply with Section 52.47(a)(1)(iv) are evaluated in Sections 20.2 to 20.5 of this report. Additional issues that the applicant considers applicable to the AP1000 design are included in DCD Tier 2 Section 1.9.4 and were evaluated by the staff.

The applicant evaluated the issues in Supplement 14 to NUREG-0933 to determine which issues were technically relevant to the AP1000 design. Their review updated the status of the items to the status in Supplement 17; however, items not relevant to the AP1000 design added between Supplements 14 and 17 were not reviewed.

The staff concludes that the applicant has adequately demonstrated compliance of the AP1000 design with 10 CFR 52.47(a)(1)(iv) in that it has addressed the issues in the relevant supplement of NUREG-0933.

20.1.2 Compliance with 10 CFR 52.47(a)(1)(ii)

As stated above, 10 CFR 52.47(a)(1)(ii) requires a design certification applicant to demonstrate compliance with any technically relevant portions of the TMI Action Plan requirements in 10 CFR 50.34(f)The applicant addressed these requirements in DCD Tier 2 Section 1.9.3 and these requirements are discussed in Section 20.6 of this report. Because of the overlap between these TMI Action Plan items and those from NUREG-0933 (discussed in Section 20.4 of this report), all the relevant 50.34(f) TMI Action Plan items are listed in Section 20.6 in tabular form. This provides the issue designation and a reference to the appropriate issue in Section 20.4 of this report which contains the evaluation of the 50.34(f) TMI Action Plan item.

The staff concludes that the applicant has adequately demonstrated compliance of the AP1000 design with 10 CFR 52.47(a)(1)(ii) in that it has addressed the relevant TMI Action Plan items in 10 CFR 50.34(f), except as noted in this report.

20.1.3 Incorporation of Operating Experience

In a staff requirements memorandum (SRM) from the Commission, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission directed the staff to ensure that the design certification process preserves operating experience insights in the certified design. The applicant submitted its evaluation for the AP1000 design in the topical report WCAP-15800, "Operational Assessment for AP1000." As discussed in Section 20.7 of this report, the staff concludes that the applicant has adequately considered operating experience in that it has addressed generic letters (GL) and bulletins issued by the Commission between January 1, 1980, and December 31, 2002, in the AP1000 design, except as noted in this report.

20.1.4 Resolution of Issues Relevant to the AP1000 Design

In DCD Tier 2 Table 1.9-2 in Section 1.9.4 in the letter dated May 28, 1993, the applicant listed the issues in Supplement 14 of NUREG-0933 that it considered relevant to the AP1000 design. The section also provides the applicant's justification for considering an issue not relevant to the design. The resolution of the issues that the applicant and the staff considered relevant to the design are discussed in Sections 20.2 through 20.6 of this report.

In Table 20.1-1, the staff lists the USIs and GSIs relevant to the AP1000 design, the sections in which these issues appear in this chapter, and the basis for the relevancy of each issue to the design. The relevancy of the issues fall into one of the following categories:

- The issue is required by 10 CFR 52.47(a)(1)(ii) or (iv) (i.e., 52.47).
- The issue was selected by the applicant as being relevant in DCD Tier 2 Section 1.9.4 (i.e., W).
- The staff decided to discuss the issue as being relevant to AP1000 (i.e., staff).

The applicant provided its justifications for considering an issue not relevant to the AP1000 design in Table 1.9-2 of DCD Tier 2 Section 1.9.4. The staff reviewed these justifications for those issues which the staff considered relevant to the design to meet 10 CFR 52.47(a)(1)(iv).

Table 20.1-1 USIs/GSIs in NUREG-0933 (Supplement 14) relevant to the AP1000 Design

Issue	Title of Issue and Section of this report	Relevancy
	Section 20.2, Task Action Plan Items	
A-1	Water Hammer	52.47/W
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	52.47/W
A-3	Westinghouse Steam Generator Tube Integrity	52.47/W
A-9	Anticipated Transient Without Scram (ATWS)	52.47/W
A-11	Reactor Vessel Materials Toughness	W
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Supports	52.47/W
A-13	Snubber Operability Assurance	52.47/W
A-17	Systems Interactions in Nuclear Power Plants	52.47/W
A-24	Qualification of Class 1E Safety-Related Equipment	52.47/W
A-25	Non-Safety Loads on Class 1E Safety-Related Equipment	52.47/W
A-26	Reactor Vessel Pressure Transient Protection	52.47/W
A-28	Increase in Spent Fuel Storage Capacity	W
A-29	Nuclear Power Plant Design for Reduction of Vulnerability to Sabotage	Staff
A-31	RHR Shutdown Requirements	52.47/W
A-35	Adequacy of Offsite Power Systems	52.47/W
A-36	Control of Heavy Loads Near Spent Fuel	52.47/W
A-40	Seismic Design Criteria Short-term Program	52.47/W
A-43	Containment Emergency Sump Performance	52.47/W
A-44	Station Blackout	52.47/W
A-46	Seismic Qualification of Equipment in Operating Plants	W
A-47	Safety Implications of Control Systems	52.47/W
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	52.47/W
A-49	Pressurized Thermal Shock	52.47/W
B-5	Ductibility of Two-Way Slabs and Shells, and Buckling Behavior of Steel Containments	52.47/W
B-17	Criteria for Safety-Related Operator Actions	52.47/W
B-22	LWR Fuel	W
B-29	Effectiveness of Ultimate Heat Sinks	W
B-32	Ice Effects on Safety-Related Water supplies	W
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for ESF Systems and Normal Ventilation Systems	52.47/W
B-53	Load Break Switch	W
B-56	Diesel Reliability	W
B-60	Loose Parts Monitoring System	Staff
B-61	Allowable ECCS Equipment Outage Periods	52.47/W
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	52.47/W
B-66	Control Room Infiltration Measurements	52.47/W
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	52.47/W
C-4	Statistical Methods for ECCS Analysis	52.47/W
C-5	Decay Heat Update	52.47/W
C-6	LOCA Heat Sources	52.47/W
C-10	Effective Operation of Containment Sprays in a LOCA	52.47/W
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	52.47/W

14	PWR Pipe Cracks	52.47/W
15	Radiation Effects on Reactor Vessel Supports	52.47/W
22	Inadvertent Boron Dilution Events	52.47/W
23	Reactor Coolant Pump Seal Failures	52.47/W
24	Automatic ECCS Switchover to Recirculation	52.47/W
29	Bolting Degradation or Failure in Nuclear Power Plants	52.47/W
43	Reliability of Air Systems	Staff
45	Inoperability of Instrumentation Due to Extreme Cold Weather	52.47/W
51	Improving the Reliability of Open-Cycle Service Water Systems	52.47/W
57	Effects of Fire Protection Systems Actuation on Safety-Related Equipment	52.47/W
67.3.3	Improved Accident Monitoring	52.47
70	PORV and Block Valve Reliability	52.47/W
73	Detached Thermal Sleeves	52.47
75	Generic Implications of ATWS Events at Salem Nuclear Plant	52.47
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	52.47/W
82	Beyond-Design-Basis Accidents in Spent Fuel Pools	Staff
83	Control Room Habitability	52.47/W
87	Failure of HPCI Steamline Without Isolation	52.47/W
89	Stiff Pipe Clamps	52.47/W
93	Steam Binding of Auxiliary Feedwater Pumps	52.47/W
94	Additional Low-Temperature Overpressure Protection for Light-Water Reactors	52.47/W
103	Design for Probable Maximum Precipitation	52.47/W
105	Interfacing System LOCA at LWRs	W
106	Piping and Use of Combustible Gases in Vital Areas	52.47/W
113	Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers	52.47/W
120	On-Line Testability of Protection Systems	52.47/W
121	Hydrogen Control for Large, Dry PWR Containments	52.47/W
122.2	Initiating Feed and Bleed	Staff
124	Auxiliary Feedwater Reliability	52.47/W
125.II.7	Reevaluation Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	Staff
128	Electric Power System Reliability	52.47/W
130	Essential Service Water Pump Failures at Multi-plant Sites	52.47/W
135	Steam Generator and Steamline Overfill	52.47/W
142	Leakage Through Electrical Isolators in Instrumentation Circuits	52.47/W
143	Availability of Chilled Water Systems and Room Cooling	52.47/W
153	Loss of Essential Service Water in LWRs	52.47
163	Multiple SG Tube Leak	52.47
168	Equipment Qualification of Electric equipment	Staff
185	Control of Recriticality following SBLOCA in PWRs	52.47
189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident	Staff
191	Assessment of Debris Accumulation on PWR Sump Performance	Staff
	Section 20.4, Three Mile Island Action Plan Items	
I.A.1.4	Long-Term Upgrade of Operating Personnel and Staffing	52.47
I.A.2.6(1)	Revise Regulatory Guide 1.8	52.47
I.A.4.1(2)	Interim Changes in Training Simulators	52.47
I.A.4.2	Long-Term Training Simulator Upgrade	52.47

I.C.1	Guidance for Evaluation and Development of Procedures for Transients and Accidents	Staff
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	52.47/W
I.C.9	Long-Term Program for Upgrading Procedures	52.47
I.D.1	Control Room Design Reviews	52.47/W
I.D.2	Plant Safety Parameter Display Console	52.47/W
I.D.3	Safety System Status Monitoring	52.47/W
I.D.5(2)	Control Room Design: Improved Instrumentation Research - Plant Status and Postaccident Monitoring	52.47/W
I.D.5(3)	Control Room Design: On-Line Reactor Surveillance Systems	52.47/W
I.F.1	Expanded Quality Assurance	52.47
I.F.2	Development of More Detailed QA Criteria	52.47/W
I.G.1	Training Requirements	W
I.G.2	Scope of Test Program	52.47
II.B.1	Reactor Coolant System Vents	52.47/W
II.B.2	Plant Shielding to Provide Postaccident Access to Vital Areas	52.47/W
II.B.3	Postaccident Sampling Capability	52.47/W
II.B.8	Rulemaking Proceedings on Degraded Core Accidents Description	52.47/W
II.D.1	Performance Testing of PWR Safety and Relief Valves	52.47/W
II.D.3	Coolant System Valves: Valve Position Indication	52.47/W
II.E.1.1	Auxiliary Feedwater System Evaluation	52.47/W
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	52.47/W
II.E.1.3	Update Standard Review Plan and Development of Regulatory Guides	52.47/W
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Staff
II.E.3.1	Pressurizer Heater Power Supply	52.47/W
II.E.4.1	Dedicated Hydrogen Penetrations	52.47/W
II.E.4.2	Containment Isolation Dependability	52.47/W
II.E.4.4	Purging	52.47/W
II.E.5.1	Design Evaluation	W
II.E.6.1	In Situ Valve Testing, Test Adequacy Study	52.47/W
II.F.1	Additional Accident Monitoring Instrumentation	52.47/W
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	52.47/W
II.F.3	Instrumentation for Monitoring Accident Conditions	52.47/W
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	52.47/W
II.J.3.1	Organization and Staffing to Oversee Design and Construction	52.47/W
II.J.4.1	Revise Deficiency Reporting Requirements	52.47
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	Staff
II.K.1(4d)	Review Operating Procedures and Training to Ensure that Operators Are Instructed Not to Rely on Level Alone in Evaluating Plant Conditions	Staff
II.K.1(5)	Safety-Related Valve Position Description	52.47
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	52.47
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementing of all Bulletin Items	52.47/W
II.K.1(16)	Implement Procedures that Identify PZR PORV "Open" Indications and that Direct Operator to Close Manually at "Reset" Setpoint	Staff
II.K.1(17)	Trip Pressurizer Level Bistable so that Pressurizer Low Pressure Will Initiate Safety Injection	52.47/W

Issue	Title and Section of this report	Relevancy
	Section 20.4, Three Mile Island Action Plan items	
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal System When Feedwater System Not Operable	W
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	52.47
II.K.1(25)	Develop Operator Action Guidelines	52.47
II.K.1(26)	Revise Emergency Procedures and Train Reactor Operators and Senior Reactor Operators	52.47
II.K.1(27)	Provide Analysis and Develop Guidelines and Procedures for Inadequate Core Cooling	52.47
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	52.47
II.K.2(10)	Hard-Wired Safety Grade Anticipatory Reactor Trip	W
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power	W
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	W
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	52.47/W
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps During LOCA	52.47/W
II.K.3(6)	Instrumentation to Verify Natural Circulation	Staff
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of Steam Generators	Staff
II.K.3(9)	Proportional Integral Derivative Controller Modification	W
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for some Event Sequences	W
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	52.47/W
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	W
II.K.3(30)	Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix K	Staff
III.A.1.2	Upgrade Licensee Emergency Support Facilities	52.47/W
III.A.3.3	Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Communication Systems	52.47/W
III.D.1.1	Primary Coolant Sources Outside the Containment	52.47/W
III.D.3.3	In-Plant Radiation Monitoring	52.47/W
III.D.3.4	Control Room Habitability	52.47/W
	Section 20.5, Human Factors Issues	
HF1.1	Shift Staffing	52.47
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	W
HF4.4	Guidelines for Upgrading Other Procedures	52.47
HF5.1	Local Control Station	52.47/W
HF5.2	Review Criteria for Human Factors Aspects of Advance Controls and Instrumentation	52.47/W

NOTES:

* 52.47: The resolution of the issue is required by 10 CFR 52.47(a)(1)(ii) and (iv).
W: Westinghouse submitted an evaluation.
Staff: The staff provided a resolution for the issue.

20.2 Task Action Plan Items

The task action plan items listed in Table 20.1-1 are evaluated against the AP1000 design in this section. The majority of the items were chosen either because (1) 10 CFR 52 (a)(1)(iv) or 10 CFR 50.34 (f) require the design to comply with them, or (2) the applicant decided that the item applied to the design and included a discussion of the item in DCD Tier 2.

Issue A-1: Water Hammer

As discussed in NUREG-0933, Issue A-1 addresses the issue of water hammer in fluid systems in nuclear power plants. Water hammer can be caused by a number of conditions, such as voiding in normally filled lines, condensation in lines, entrainment of water in steam-filled lines, or rapid valve actuation. Issue A-1 addresses these probable causes, as well as possible methods for minimizing the susceptibility of systems to water hammer through design and operational considerations. This issue was resolved with the publication of NUREG-0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," Revision 1, dated March 1984, which contained evaluation results of water hammer events, as well as details of recommendations and measures for water hammer prevention and mitigation.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that the AP1000 design meets the guidance of applicable standard review plan (SRP) sections in NUREG-0800 that provide criteria for mitigation of water hammer concerns and NUREG-0927, and addressed design features and system operation that mitigate or prevent water hammer damage. The applicant stated that design features are incorporated in the applicable systems, including the steam generator (SG) feedrings and piping, passive core cooling system, passive residual heat removal system, service water system, feedwater system, and steamlines. These features are summarized below.

The automatic depressurization system uses multiple, sequenced valve stages to provide a relatively slow, controlled depressurization of the reactor coolant system (RCS), which helps reduce the potential for water hammer. Once depressurization is complete, gravity injection from the refueling water storage tank is initiated by opening check valves, which reposition slowly. Gravity injection flow actuates slowly, without water hammer, as the pressure differential across the check valves equalizes, and the valves open and initiate flow.

The passive residual heat removal system exchangers are normally aligned with open inlet valves and closed discharge valves. This keeps the system piping at RCS pressure and prevents water hammer upon initiation of flow through the heat exchangers.

The core makeup tanks are normally aligned to the cold leg to keep the tanks at RCS pressure. The line is also normally kept filled with steam to prevent water hammer upon actuation of the core makeup tank. DCD Tier 2 Section 6.3 provides additional information on the passive core cooling system.

The potential for water hammer in the feedwater line is minimized by the design and operation of the feedwater delivery system. The SG features include introducing feedwater into the SG at

an elevation above the top of the tube bundles and below the normal water level by a top discharge spray tube feedring. The layout of the feedwater line is consistent with industry standard recommendations to reduce the potential of a SG water hammer. In addition, operational limitations on flow to recover SG levels and on early feedwater flow into the SG minimize the potential for water hammer.

The startup feedwater system is a non-safety-related system that provides heated feedwater during plant startup, shutdown, and hot standby. The heated feedwater reduces the potential for water hammer in the feedwater piping and SG feedrings.

The main steamlines are designed to remove accumulated condensate from the main steamlines and to maintain the turbine bypass header at operating temperature during plant operation. The system is designed to accommodate flows during startup, shutdown, transients, and normal operation. This is to protect the turbine and turbine bypass valves from water slug damage.

The above discussions, supplemented by the various measures to minimize the potential of water hammer described in DCD Tier 2 Sections 1.9.4.2.2, 3B.2.3, 5.4.6, 6.3.2.5, 5.4.2.2, 5.4.7.2, 9.2.1.2.2, 10.4.7, and 14; and in the Topical Report WCAP-15799, “AP1000 Compliance with the SRP Acceptance Criteria,” provide acceptable commitments for the AP1000 design to meet water hammer-related guidelines in applicable sections of the SRP and NUREG-0927.

The results from a small-break loss-of-coolant accident (SBLOCA) test performed earlier for AP600 at Oregon State University have indicated that rapid condensation events have the potential to cause unanticipated dynamic loads in the RCS. The staff has concluded that these results are applicable to the AP1000 design. The staff's evaluation of these test results found that the loads so induced are small and inconsequential to components and piping integrity. Based on review of this information, the staff concludes that Issue A-1 is resolved for the AP1000 design.

Issue A-2: Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

As discussed in NUREG-0933, Issue A-2 addresses the concerns raised in 1975 by Virginia Electric Power Company that an asymmetric loading on the reactor vessel supports resulting from a pipe break at the vessel nozzle had not been considered by the utility or the applicant in the original design of the reactor vessel support system for North Anna Units 1 and 2. In the postulated event at the vessel nozzle, asymmetric loss-of-coolant accident (LOCA) loads could result from forces induced on the reactor internals by transient differential pressures across the core barrel, and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the more detailed analytical models, it became apparent to the applicant that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports.

The issue was resolved with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981. The asymmetric loads on the reactor vessel, internals,

primary coolant loop, and components should not exceed the limits imposed by the applicable codes and standards. The staff also issued GL 84-04, "Safety Evaluation of the applicant Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," on February 1, 1984, to permit the application of leak-before-break (LBB) technology to eliminate the dynamic effects from a postulated pipe rupture from the design basis. Subsequently, the staff revised general design criteria (GDC) 4 to permit the application of LBB.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that the use of mechanistic pipe break (or LBB) criteria permits the elimination of the evaluation of dynamic effects of pipe breaks in the analysis of structures, systems, and components (SSCs). GDC 4 allows the use of LBB to eliminate from the design basis the dynamic effects of pipe ruptures postulated at locations defined in DCD Tier 2 Section 3.6.2. The 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 main reactor coolant loops are designed in accordance with LBB criteria. This is described in DCD Tier 2 Section 3.6.3 and Appendix 3B.

The staff review of this information is contained in Section 3.6.3 of this report and resulted in several open items; therefore, USI A-2 is resolved for the AP1000 design pending resolution of these open items.

Issue A-3: Westinghouse SG Tube Integrity

As discussed in NUREG-0933, Issue A-3 addresses staff concerns related to (SG) tube degradation. These concerns stemmed from the fact that the SG tubes are a part of the RCS boundary, and that tube ruptures allow primary coolant into the secondary system where its isolation from the environment is not fully ensured. In 1978, Issues A-3, A-4, and A-5 were established to evaluate the safety significance of tube degradation in the applicant, Combustion Engineering (CE), and Babcock and Wilcox (B&W) SGs, respectively. These studies were later combined into one effort because of the similarity of many problems among the pressurized-water reactor (PWR) vendors.

This issue was resolved and no new requirements were established (U.S. NRC, SECY-88-272, “Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding SG Tube Integrity,” September 1988). However, the staff issued GL 85-02, “Staff Recommended Issues Regarding SG Tube Integrity,” dated April 17, 1985, to provide recommended actions from NUREG-0844.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that the AP1000 SGs are designed in accordance with GL 85-02 and NUREG-0844. The SGs have features described in DCD Tier 2 Section 5.4.2 to enhance tube performance and reliability. These features include the following:

- The design provides access to all tubes to perform inservice inspection (ISI)

- The tubes are fabricated from thermally treated nickel-chromium-iron Alloy 690.

- Support to the tubes is provided by stainless steel support plates

- Contact between tubes and support plates is by the trifoil tube hole design, which provides a high sweeping velocity to reduce sludge accumulation in crevices.

- The portion of the tube within the tubesheet is fully expanded to close the crevices between the tube and tubesheet.

- The SG channel head is designed to facilitate the replacement of the SG, if this is required.

As discussed in DCD Tier 2 Sections 5.2.4 and 5.4.2, the development of the SG tube preservice inspection (PSI) and inservice inspection (ISI) programs is the responsibility of the COL applicant. SG tube integrity is verified in accordance with this surveillance program as discussed in DCD Tier 2 Section 5.4.15. The programs are plant specific and will be reviewed by the staff individually for each license application referencing the AP1000 design certification against the staff's regulatory criteria in place at the time of its review. This action item is designated as COL Action Item 20.2-1.

The staff concludes that Issue A-3 is resolved for the AP1000 design.

Issue A-9: Anticipated Transient Without Scram

As discussed in NUREG-0933, Issue A-9 addressed the issue of ensuring that the reactor can attain safe shutdown after incurring an anticipated transient with a failure of the reactor trip system (RTS). An anticipated transient without scram (ATWS) is an expected operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power (LOOP) to the reactor) that is accompanied by a failure of the RTS to shut down the reactor.

Generic Safety Issue A-9 was resolved with the publication of 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) for Light-Water-Cooled Reactors."

The acceptance criteria for the resolution of Issue A-9 are as follows:

- Compliance with the mitigation requirement of 10 CFR 50.62(c)(1) that plant equipment must automatically initiate emergency feedwater (EFW) and turbine trip under conditions indicative of an ATWS. This equipment must function reliably and must be diverse and independent from the RTS.

- Compliance with the prevention requirement of 10 CFR 50.62(c)(2) that the plant must have a scram system that is diverse and independent from the existing RTS.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that the AP1000 design complies with the requirements in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated

Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," except that the AP1000 does not have a safety-related auxiliary feedwater system, and includes a discussion of the design features to minimize the probability of an ATWS in DCD Tier 2 Subsections 1.9.5.1.3 and 7.7.

The applicant indicates that the AP1000 design complies with the requirements of 10 CFR 50.62 with a diverse actuation system that includes the AMSAC (ATWS mitigation system actuation circuitry) protection features mandated by 10 CFR 50.62 by tripping the turbine and diversely actuating selected engineered safeguards functions.

There are other AP1000 design features aimed at minimizing the probability of ATWS occurrence and mitigating the consequences, as discussed in DCD Subsection 1.9.5.1.3. For the AP1000 design with passive core cooling systems, the staff requires that an ATWS analysis be performed to demonstrate that its ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plant designs. In response to request for additional information (RAI) 440.014, Revision 1 (the applicant letter DCP/NRC1558, March 28, 2003), the applicant provided the analysis of a complete loss of normal feedwater without reactor trip using the LOFTRAN code.

The detailed discussion of this issue is found in section 15.2.9 of this report. The staff reviewed the AP1000 design and analyses, and concluded that the AP1000 design meets the intent of 10 CFR 50.62 requirements. The staff, therefore, concludes that Issue A-9 is resolved for the AP1000 design.

Issue A-11: Reactor Vessel Material Toughness

In DCD Tier 2 Table 1.9-2, the applicant identifies that it considers Issue A-11 relevant to the AP1000 design; however, this issue is not required for the AP1000 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue A-11 addresses the NRC concern that, because of the remote possibility of failure of nuclear reactor pressure vessels designed to the American Society of Mechanical Engineers (ASME) Code, the design of nuclear facilities must provide protection against reactor vessel failure.

Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. As plants accumulate more service time, neutron irradiation reduces the material fracture toughness and initial safety margins. This issue was resolved with the publication of NUREG-0744, "Resolution of the Task A-11, Reactor Vessel Materials Toughness Safety Issue," Revision 1, October 1982, and GL 82-26, "NUREG-0744, Revision 1, Pressure Vessel Material Fracture Toughness," November 12, 1982. This issue did not result in establishing new regulatory requirements.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that the AP1000 reactor vessel design complies with the requirements of Appendix G, "Fracture Toughness Requirements," of 10 CFR

Part 50 and includes features to reduce neutron fluence, enhance material toughness at low temperature, and eliminate weld seams in critical areas. Material requirements are discussed in DCD Tier 2 Sections 5.3.1 and 5.3.2, and pressure and temperature limits are provided in DCD Tier 2 Section 5.3.3.

The AP1000 reactor vessel design complies with the requirements of 10 CFR Part 50, Appendix G, and includes various features for the vessel to reduce neutron fluence, enhance material toughness at low temperatures, and eliminate weld seams in critical areas. The staff evaluation of the vessel material properties and fracture toughness is provided in Sections 5.3.2, 5.3.3, 5.3.4, and 5.3.5 of this report.

The staff concludes that Issue A-11 is resolved for the AP1000 design.

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

During the course of the licensing action for North Anna, Units 1 and 2, a number of questions were raised about the potential for lamellar tearing and low-fracture toughness of the SG and reactor coolant pump (RCP) support materials for these facilities. Concerns regarding the supports at North Anna were applicable to all PWRs. This was designated as Issue A-12 in NUREG-0933.

This issue was resolved and no new requirements were established (NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," Revision 1, October 1983). However, the staff recommended developing guidance for new plants on the basis of the fracture toughness requirements of Subsection NF of Section III of the ASME Code.

Westinghouse describes the SG and RCP supports in DCD Tier 2 Section 5.4.10. The supports are designed and fabricated in accordance with Subsection NF of Section III of the ASME Code. Westinghouse states that Subsection NF requirements provide acceptable fracture toughness for the support materials.

The staff concludes that the Westinghouse response to Issue A-12 addresses the structural integrity of SG and RCP supports. Therefore, Issue A-12 is resolved for the AP1000 design.

Issue A-13: Snubber Operability Assurance

Snubbers are primarily used as seismic and pipe whip restraints at nuclear power plants. They function as rigid supports for restraining the motion of attached systems or components under such rapidly applied load conditions as earthquakes, pipe breaks, and severe hydraulic transients, while allowing free thermal expansion of the piping systems and components during various operating conditions. Issue A-13 in NUREG-0933 addressed the concern of a substantial number of snubber malfunctions, the most frequent of which were (1) seal leakage in hydraulic snubbers, and (2) high rejection rate during functional testing of snubbers. This issue has been resolved and new guidelines were established in 1981, with the revision of SRP

Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

The staff review of DCD Tier 2 Section 3.9.3.4.3 concludes that the information provided is consistent with the guidelines in SRP 3.9.3 relative to snubber operability, and provides an acceptable approach to address the issue of snubber operability. The staff review of this issue is included in Section 3.9.3.3 of this report. On the basis of this evaluation, the staff concluded that the guidelines in SRP 3.9.3 relative to snubber operability have been met, and that Issue A-13 is resolved for the AP1000 design.

Issue A-17: Systems Interactions in Nuclear Power Plants

As discussed in NUREG-0933, Issue A-17 addressed concerns regarding adverse systems interactions (ASIs) in nuclear power plants. Depending on how they propagate, ASIs can be classified as functionally coupled, spatially coupled, and induced-human-intervention coupled. As discussed in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17," dated August 1989, and GL 89-18, "Resolution of Unresolved Safety Issue A-17, Systems Interactions in Nuclear Power Plants," dated September 6, 1989, Issue A-17 concerns ASIs caused by water intrusion, internal flooding, seismic events, and pipe ruptures.

A nuclear power plant comprises numerous SSCs that are designed, analyzed, and constructed using many different engineering disciplines. The degree of functional and physical integration of these SSCs into any single power plant may vary considerably. Concerns have been raised about the adequacy of this functional and physical integration and the coordination process. The Issue A-17 program was initiated to integrate the areas of systems interactions and consider viable alternatives for regulatory requirements to ensure that the ASIs have been or will be minimized in operating plants and new plants. Within the framework of the program, the staff requested, as stated in NUREG-0933, that plant designers consider the operating experience discussed in GL 89-18 and use the probabilistic risk assessment (PRA) required for future plants to identify the vulnerability and reduce ASIs.

This issue identified the need to investigate the potential that unrecognized subtle dependencies, or systems interactions, among SSCs in a plant could lead to safety significant events. In NUREG-1174,"Evaluation of Systems Interactions in Nuclear Power Plants: Technical Findings Related to Unresolved Safety Issue A-17," dated May 1989, intersystem dependencies are categorized on the basis of the way they propagate into functionally-coupled, spatially-coupled, and induced human-intervention coupled systems interactions. The occurrence of an actual ASI or the existence of a potential ASI, as well as the potential overall safety impact, is a function of an individual plant's design and operational features. For the AP1000 with new or differently configured passive and active systems, a systematic search for ASIs is necessary.

In DCD Tier 2 Section 1.9.4.2.2, Westinghouse stated that the AP600 was the subject of a systematic evaluation of potential adverse systems interactions documented in WCAP-14477, "The AP600 Adverse System Interaction Evaluation Report, and that the conclusions of WCAP-14477 are applicable to the AP1000 since the fluid system design for the AP1000 is the same

as the AP600. However, in response to a staff RAI, the applicant submitted WCAP-15992, “AP1000 Adverse System Interactions Evaluation Report,” dated November 2002, and Revision 1, dated February 2003.

The purpose of the report was to identify possible adverse interactions among safety-related systems and between safety-related and non-safety-related systems, and to evaluate the potential consequences of such interactions. The staff reviewed this issue as part of the regulatory treatment of non-safety systems (RTNSS) described in Chapter 22 of this report.

The staff concludes that the applicant has adequately assessed possible ASIs and their potential consequences in WCAP-15992, Revision 1. Therefore, Issue A-17 is resolved for the AP1000 design.

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

Construction permit (CP) applicants for which safety evaluation reports were issued after July 1, 1974, were required by the NRC to qualify all safety-related equipment to Institute of Electrical and Electronics Engineers (IEEE)-323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." From the time this standard was originated, the industry developed methods that were used to qualify equipment in accordance with the standard. To assess the adequacy of the equipment qualification methods and acceptance criteria used by nuclear steam supply system (NSSS) and balance-of-plant (BOP) vendors, the NRC determined that a generic approach was required. This was designated as Issue A-24 in NUREG-0933 and was resolved with the publication of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," dated July 1981.

In DCD Tier 2 Section 1.9.4, the applicant states that the AP1000 environmental qualification methodology described in DCD Tier 2 Appendix 3D is founded on the generic Westinghouse qualification program approved by the NRC. The applicant also states that this methodology addresses the requirements of GDC 4 and 10 CFR 50.49, as well as the guidance of Regulatory Guide (RG) 1.89 and IEEE Standard 323-1974.

On the basis of the staff’s review, which is discussed in Section 3.11 of this report, the staff concludes that the applicant’s approach to environmental qualification of Class 1E equipment is in compliance with 10 CFR 50.49 and Issue A-24 is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue A-25 addressed a review of whether non-safety-related loads should also be allowed to share Class 1E power sources. The Class 1E power sources provide the electric power for the plant systems that are essential to reactor shutdown, containment isolation, reactor core cooling, containment heat removal, and preventing significant release of radioactive material to the environment. As discussed in NUREG-0933, this issue was resolved in Revision 2 to RG 1.75, "Physical Independence of Electric Systems," with minor exceptions (see Section 8.3.2.3).

The 125 Vdc emergency lighting in the main control room and in the remote shutdown area is non-Class 1E and is fed from a Class 1E Uninterrupted Power Supply (UPS) through two series fuses that are coordinated for isolation. Present regulatory practice allows the connection of non-safety loads to Class 1E (emergency) power sources if it can be shown that the connection of non-safety loads will not result in degradation of the Class 1E system. In the AP1000 design, either of these fuses is able to interrupt any fault current before initiation of a trip of any upstream fuse. No credible failure of non-Class 1E equipment or systems will degrade the Class 1E system below an acceptable level.

Therefore, Issue A-25 is resolved for the AP1000 design.

Issue A-26: Reactor Vessel Pressure Transient Protection

Since 1972, there have been many reported pressure transients that have exceeded the pressure-temperature limits specified in technical specifications (TS) for PWRs. The majority of these events occurred at relatively low reactor vessel temperatures, at which the material has less toughness and is more susceptible to failure through brittle fracture. This is Issue A-26 in NUREG-0933, which was resolved with the issuance of SRP Section 5.2.2, "Overpressure Protection." Applicants for construction permits and operating licenses were requested to design an overpressure protection system for light-water reactors (LWRs) following the guidance provided in SRP Section 5.2.2.

In DCD Tier 2 Subsection 1.9.4.2.2, the applicant states that the AP1000 design conforms to the criteria in Branch Technical Position (BTP) RSB 5-2, "Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures," of SRP Section 5.2.2. The pressurizer is sized to accommodate most pressure transients, and overpressure protection for the RCS is provided by either the pressurizer safety valves during power operation, or the normal residual heat removal relief valve for low-temperature overpressure protection, as described in DCD Tier 2 Subsection 5.2.2. The staff provides its evaluation of the RCS overpressure protection in Section 5.2.2 of this report.

The staff concludes that the AP1000 design satisfies the BTP RSB 5-2 requirements and, therefore, considers Issue A-26 resolved for the AP1000 design.

Issue A-28: Increase in Spent Fuel Pool Storage Capacity

The applicant identifies, in DCD Tier 2 Table 1.9-2, that it considers Issue A-28 relevant to the AP1000 design; however, this issue is not required for the AP1000 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

Issue A-28 of NUREG-0933 addressed the development of consistent and formalized acceptance criteria regarding the conversion of existing spent fuel storage pools to higher density storage racks, to increase storage capacity. This issue was resolved with the NRC letter to licensees on April 17, 1978, which provided in a single document, the criteria used by the staff to evaluate applications for spent fuel pool storage modifications.

In DCD Tier 2 Section 1.9.4, the applicant states that the AP1000 design incorporates the NRC criteria and the heat load is evaluated for the stated spent fuel storage capacity.

The staff evaluated the conformance of the AP1000 spent fuel pool design to the NRC criteria in Section 9.1.2 of this report and, on the basis of the staff's conclusions in this section, Issue A-28 is resolved for the AP1000 design.

Issue A-29: Nuclear Power Plant Design for Reduction of Vulnerability to Sabatoge

This issue has not been reviewed by the staff and, therefore, as per Section 13.6, it is in an Open Item 13.6-1.

Issue A-31: Residual Heat Removal Shutdown Requirements

As discussed in NUREG-0933, Issue A-31 addressed the ability to transfer heat from the reactor to the environment after shutdown, which is an important safety function. It was resolved in 1978 with the issuance of SRP Section 5.4.7, "Residual Heat Removal (RHR) System."

The safe shutdown of a nuclear power plant following an accident not related to a LOCA has typically been interpreted as achieving "hot-standby" condition. The NRC has placed considerable emphasis on the hot-standby condition of a power plant in the event of an accident or other abnormal occurrence and, similarly, on long-term cooling, which is typically achieved by the RHR system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than the hot-standby-condition values. Even though it may generally be considered safe to maintain a reactor in hot-standby condition for a long time, experience shows that certain events have occurred that required eventual cooldown or long-term cooling until the RCS is cold enough for personnel to inspect the problem and repair it.

In DCD Tier 2 Subsection 1.9.4.2.2, the applicant states that the AP1000 design includes passive safety-related core decay heat removal systems that establish and maintain the plant in a safe-shutdown condition following design-basis events, and it is not necessary that these passive systems achieve cold shutdown as defined in RG 1.139.

The passive core cooling system is designed to maintain plant safe-shutdown conditions indefinitely. Cold-shutdown condition is necessary only to gain access to the RCS for inspection, maintenance, or repair. For the AP1000 design, cold-shutdown conditions can be achieved using highly reliable, but non-safety-related systems, which have similar redundancy as current generation safety-related systems and are supplied with alternating current (ac) power from either onsite or offsite sources. The non-safety related normal RHR system (RNS) is discussed in DCD Subsection 5.4.7. The staff provides its evaluation of the RNS in Section 5.4.7 of this report.

The applicant states that the passive RHR system, whose capability is discussed in DCD Subsection 6.3, can achieve hot-standby conditions immediately and can reduce the reactor

coolant temperature to 215.6 °C (420 °F) within 36 hours. The reactor pressure is controlled and can be reduced to 1.72 MPa (250 psig). The passive RHR system also provides a closed cooling system to maintain long-term cooling. Therefore, the AP1000 complies with GDC 34 by using a more reliable and simplified system for both hot-standby and long-term cooling modes, and it is not necessary that these passive systems achieve cold shutdown as defined by RG 1.139.

In GDC 34, the NRC requires a RHR to be provided with suitable redundancy in components and features to assure that, with or without onsite or offsite power, it can accomplish its safety functions so that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. No definition is specified as the safe-shutdown condition for which the RHR system should accomplish this. The Electric Power Research Institute (EPRI) Utility Requirements Document (URD) proposed that the safe-shutdown condition be defined as 215.6 °C (420 °F) for the passive Advanced Light Water Reactor (ALWR) designs. The staff concluded that cold-shutdown is not the only safe stable shutdown condition that can maintain the fuel and reactor pressure boundary within acceptable limits. In SECY-94-084, Section C, "Safe Shutdown Requirements," the staff recommended, and the Commission approved, that the EPRI-proposed 215.6 °C (420 °F) criteria or below, rather than the cold-shutdown condition required by RG 1.139, be accepted as a safe stable condition, which the passive RHR system must be capable of achieving and maintaining following non-LOCA events. This acceptance is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of RTNSS. The SECY paper also states that the passive safety system capabilities can be demonstrated by appropriate evaluations during detailed design analyses, including the following:

(1) A safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition, that no transients will result in the specified acceptable fuel design limits and pressure boundary design limit being violated, and that no high-energy piping failure being initiated from this condition will result in violation of 10 CFR 50.46 criteria.

(2) A probabilistic reliability analysis, including events initiated from the safe-shutdown conditions, to ensure conformance with the safety goal guidelines. The PRA would also determine the reliability/availability missions of risk-significant systems and components as a part of the effort for RTNSS.

The staff discusses the performance of the passive system capability in Chapters 6 and 15 of this report and the RTNSS issue regarding the availability of the RNS system during shutdown and refueling conditions in Subsection 19.3, and found them acceptable for AP1000 design. Therefore, based on the discussion, the staff considers Issue A-31 resolved for the AP1000 design.

Issue A-35, "Adequacy of Offsite Power Systems"

In GDC 17 of Appendix A to 10 CFR Part 50, the NRC requires that an offsite electric power system be available to assure that (1) the fuel and reactor boundary are maintained within

specified acceptable limits, and (2) core cooling, containment integrity, and other vital safety functions are maintained during accident conditions.

The AP1000 design includes an offsite power source; however, the AP1000 design does not require any offsite ac power source to achieve and maintain safe shutdown and, therefore, this issue is not applicable to the AP1000 design.

Therefore, Issue A-35 is not applicable for the AP1000 design.

Issue A-36: Control of Heavy Loads Near Spent Fuel

At all nuclear plants, overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, radioactivity could be released to the environment. Such an occurrence would also have the potential for overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases to the environment could exceed the exposure guidelines of 10 CFR Part 100. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there was a need for a systematic review of requirements, facility designs, and TS regarding the movement of heavy loads to assess safety margins and improve them where necessary. This was designated as Issue A-36 in NUREG-0933.

The issue was resolved with the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," dated July 1980, and SRP Section 9.1.5, "Overhead Heavy Load Handling Systems."

In DCD Tier 2 Section 1.9.4, the applicant states that the AP1000 design conforms to NUREG-0612 and Section 9.1.5 of the SRP. The light-load handling systems are described in DCD Tier 2 Section 9.1.4 and the overhead heavy-load handling systems are described in DCD Tier 2 Section 9.1.5.

The staff evaluated the conformance of the AP1000 design to NUREG-0612 and Section 9.1.5 of the SRP in Sections 9.1.4 and 9.1.5 of this report and, on the basis of the staff's conclusions in these sections, Issue A-36 is resolved for the AP1000 design.

Issue A-40: Seismic Design Criteria Short Term Program

As discussed in NUREG-0933, Issue A-40 addressed short-term improvements in seismic design criteria. The objectives of Issue A-40 were the following:

- Investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites
- Investigate alternative approaches, where desirable
- Quantify the overall conservatism of the design sequence
- Modify the NRC criteria in the SRP, where justified

This issue was initiated in 1978 to identify and quantify conservatism in the seismic design process, and to develop a basis for revising SRP Section 3.7 on seismic design analyses.

To resolve this issue, the staff revised SRP Sections 2.5.2, "Vibratory Ground Motion," 3.7.1, "Seismic Design Parameters," 3.7.2, "Seismic System Analysis," and 3.7.3, "Seismic Subsystem Analysis," to address areas of vibratory ground motion; design time-history criteria; development of floor response criteria, damping values, and soil-structure interaction (SSI) uncertainties; and combination of modal responses. The revisions also addressed seismic analysis of the above-ground tanks and Category 1 buried piping. The revised SRP Section 3.7 provided guidelines for the (1) site-specific ground response spectra, (2) justification of the use of single synthetic ground motion time-history by power spectral density function, (3) basis for location and limitation of input ground motion reduction for SSI analysis, and (4) design of above-ground vertical tanks and buried piping.

In DCD Tier 2 Sections 2.5.2.1 and 2.5.4.6, the applicant stated that the COL applicant referencing the AP1000 design will perform site-specific evaluation, and demonstrate the acceptability of the AP1000 design to the site-specific characteristics. On the basis of its evaluation discussed in Sections 2.5.2 and 2.5.4 of this report, the staff concludes that performance of site-specific evaluations of seismic and geotechnical characteristics of the site by the COL applicant is acceptable.

An acceptable resolution of Issue A-40 is that future nuclear power plants should conform to the seismic design guidance of Revision 2 to SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. The AP1000 response to Issue A-40 in DCD Tier 2 Section 1.9.4.2.2 references the criteria and methodology described in DCD Tier 2 Section 3.7 as the basis for resolving this issue. The staff's review of DCD Tier 2 Section 3.7 is discussed in Sections 3.7.1, 3.7.2, and 3.7.3 of this report. On the basis of its evaluations in these sections, the staff concludes that the AP1000 design is consistent with the guidelines in Revision 2 of SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Therefore, Issue A-40 is resolved for the AP1000 design.

Issue A-43: Containment Emergency Sump Performance

Generic Issue A-43 concerns the availability of adequate cooling water following a LOCA when long-term recirculation from the PWR containment sump or boiling water reactor (BWR) emergency care cooling system (ECCS) suction intake is required to provide core cooling. The recirculation cooling water must be sufficiently free of LOCA-generated debris and ingested air so that pump performance is not impaired, thereby degrading long-term recirculation flow capability. Further information concerning Issue A-43 and its resolution may be found in GL 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage."

The staff's evaluation of the adequacy of the in-containment refueling water storage tank (IRWST) and containment recirculation screens is in Section 6.2.1.8 of this report. On the basis of the staff's evaluation, this issue is considered to be resolved for the AP1000 design

because the applicant adequately addressed the sump performance concerns identified by the staff in connection with Issue A-43.

Issue A-44: Station Blackout (SBO)

Generic Issue A-44 was resolved with the publication of 10 CFR 50.63, which provides requirements that LWRs be able to withstand for a specified duration and recover from a SBO. It addresses the likelihood of the loss of all ac power at the site, and the potential for severe core damage after the SBO.

In DCD Tier 2, the applicant stated that ac electrical power is not needed to establish or maintain a plant safe-shutdown condition for the AP1000 design. But, the design includes two redundant, non-Class 1E diesel generators to provide electrical power for non-safety-related active systems that provide a defense-in-depth function. The non-Class 1E diesel generators are identified as risk-significant in the scope of the design-reliability assurance program DCD Tier 2 Section 16.2, to the NRC on October 3, 1996. Table 16.2-1, "Risk Significant SSCs Under the Scope of D-RAP" lists non-Class 1E diesel generators as RTNSS important. The RTNSS issue is resolved in Section 8.5.2.4 of this report, therefore, Issue A-44 is resolved for the AP1000 design.

Issue A-46 Seismic Qualification of Equipment in Operating Plants

Issue A-46, of NUREG-0933, addressed the need to establish an explicit set of guidelines to verify the seismic adequacy of mechanical and electrical equipment at older operating plants instead of backfitting the current design criteria for new plants. Requirements for resolution of this issue were included in GL 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Plants, Unresolved Safety Issue (USI) A-46," on February 19, 1987.

The AP1000 response to Issue A-46 in DCD Tier 2 Section 1.9.4.2.2 states that this issue is applicable to operating plants, and not to plants to be constructed. Therefore, Issue A-46 does not apply to the AP1000, which is designed in accordance with current seismic qualification (not verification) requirements. DCD Tier 2 Section 1.9.4.2.2 also stated that the seismic Category 1 mechanical and electrical equipment in the AP1000 design will be qualified in accordance with the AP1000 qualification methodology discussed in DCD Tier 2 Section 3.10 . The staff review of this seismic qualification methodology is included in Section 3.10 of this report. Based on review of this information, the staff agrees that Issue A-46 is not applicable to the AP1000 design.

Issue A-47: Safety Implications of Control Systems

As discussed in NUREG-0933, Issue A-47 concerns the potential for accidents or transients becoming more severe as a result of control systems failures, including power supply faults. Within this issue, the staff performed an in-depth review of non-safety-related control systems and assessed the effect of control system failures on plant safety.

Non-safety-grade control systems are not relied on to perform any safety functions, but they are used to control plant processes that could have a significant impact on plant dynamics. For the resolution of Issue A-47, the NRC evaluated the effects of control system failures on PWR reference plants, including a design subjected to single and multiple control system failures during automatic and manual modes of operation. The staff's two concerns related to the design were: (1) SG overfill and (2) reactor core heat removal to cold shutdown after a small-break LOCA, without overcooling the reactor vessel. The NRC issued GL 89-19, "Request for Action Related to Resolution of USI A-47, Pursuant to 10 CFR 50.54(f)," dated September 20, 1989, which required all operating PWR plants and plants under construction to provide the following:

- automatic protection from SG overfill by the main feedwater system (MFWS) and separate from the MFWS control system

- plant procedures and TS surveillance requirements to periodically verify the operability of the overfill protection during power operation

The resolution of Issue A-47 is that the plant shall have, as a minimum, control-grade protection against SG overfill by the MFWS, and TS and plant operating procedures to ensure in-service verification of the availability of the overfill protection, in accordance with GL 89-19.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that, for the AP1000 design, control system failures are considered as potential initiating events. The analyses of transients resulting from these failures demonstrated that the consequences are bounded by American Nuclear Society (ANS) Condition II criteria and no design-basis failure for a control system is expected to violate this criteria.

The integrated control system for the AP1000 design was stated to obtain certain of its 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 this 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 being used in the control system. This capability exists for bypassed sensors or for sensors whose signals diverge from the expected error tolerance.

The AP1000 plant control system is stated to incorporate design features as redundancy, automatic testing, and self-diagnostics to prevent challenges to the protection and safety monitoring systems. DCD Tier 2 Chapter 7 provides a discussion of the AP1000 instrumentation and controls.

In DCD Tier 2 Sections 7.2.1.1.6, 7.3.1.2.6, 7.7.1.8 and Figure 7.2-1, sheet 10, the applicant addresses feedwater isolation function (SG overfill protection). The protection is provided by a safety-grade SG high-water-level (High-2) signal with a two-out-of-four initiating logic. The

plant control system uses a lower SG water level setpoint, High-1, to close the feedwater control valves. This provides an interval for operator action to prevent total isolation of the SG and reactor trip before the safety-grade High-2 setpoint is exceeded. The safe-grade signal closes the MFWS control valves and isolation valves. This is provided in the reactor trip system logic, which is sufficiently separated from the MFWS control system. The AP1000 TS (DCD Tier 2 Chapter 16), TS 3.3.1, "Reactor Trip System Instrumentation," and TS 3.7.3, "Main Feedwater Isolation and Control Valves," provide requirements that adequately address the surveillance requirements to verify the operability of the SG overfill protection. Therefore, the staff concludes that the Issue A-47 is resolved for the AP1000 design.

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

This issue remains open because DCD Tier 2 for the control of combustible gas in containment during accidents does not comply with current regulations.

The NRC has proposed major changes to 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,” and related changes to 10 CFR 50.34 and 10 CFR 52.47, along with the creation of a new rule, 10 CFR 50.46a (see 67 FR 50374, August 2, 2002). These proposed changes are meant to risk-inform the combustible gas control requirements, and constitute significant relaxations of the requirements. The staff plans to finalize the rule changes during 2003.

DCD Tier 2 is written in anticipation of these rule changes. As such, it is not in compliance with the current, more-restrictive regulations. Furthermore, until the proposed rule changes are final and effective, the staff cannot know for certain if the DCD will comply with the revised rule. Therefore, the issue of containment combustible gas control, as well as Issue A-48, must remain open at this time.

This is an Open Item 6.2.5-1.

Issue A-49: Pressurized Thermal Shock

The issue of pressurized thermal shock arises in PWRs because unanticipated transients or design-basis postulated accidents could result in severe overcooling (thermal shock) of the reactor pressure vessel concurrent with or followed by repressurization. In these events, rapid cooling of the internal surfaces of the reactor vessel results in thermal stresses with a maximum thermal tensile stress at the inside surface. The magnitude of the thermal stress depends on the temperature profile across the vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stress if the vessel is pressurized.

As discussed in NUREG-0933, Issue A-49 addressed the concern that neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel’s inner surface, that crack could grow to a size that might threaten vessel integrity. The staff’s concern is the possibility of vessel failure as a result of a severe

pressurized overcooling event, or pressurized thermal shock (PTS). As long as the fracture toughness of the reactor vessel material is relatively high, such events are not expected to cause vessel failure. However, the fracture toughness decreases during the operating life of a nuclear power plant from the fast neutron flux. The rate of decrease is dependent on the chemical composition of the material and the amount of irradiation. If the fracture toughness has been reduced significantly, severe high pressure-low temperature events could cause propagation of small flaws that could exist near the inner surface of the vessel. The assumed initial flaw might propagate into a crack through the vessel wall to threaten vessel integrity and core cooling capability.

This issue was resolved and new requirements were established in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." The rule establishes screening criteria that are related to the fracture toughness of the reactor vessel. The risk from pressure and temperature (P/T) events is acceptably low for reactor vessel materials that are projected to be below the PTS screening criteria.

In DCD Tier 2 Section 1.9.4.2.2, applicant states that the AP1000 design complies with the requirements of 10 CFR 50.61. Reactor vessel integrity for the AP1000 design is discussed in DCD Tier 2 Section 5.3.4.

The staff evaluation of this issue discussed in Section 5.3.4 of this report concluded that the reactor vessel beltline materials proposed for the AP1000 design are projected to be below the screening criteria in 10 CFR 50.61. Compliance with this rule is an acceptable basis for resolving this issue. Therefore, Issue A-49 is resolved for the AP1000 design.

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

In NUREG-0933, this issue was divided into the following two parts, which were evaluated separately:

Part I – Ductility of Two-Way Slabs and Shells

Part I of Issue B-5 was defined in NUREG-0471, "Generic Task Problem Descriptions," dated June 1978, and addressed the lack of information related to the behavior of two-way reinforced-concrete slabs loaded dynamically in biaxial tension, flexure, and shear. The objective was to develop design requirements for concrete two-way slabs to resist loading caused by a LOCA or high-energy line break (HELB). An acceptable resolution to this issue is to apply the two-way reinforced-concrete slab analysis methods to adequately address dynamic loading in biaxial membrane tension, flexure, and shear due to a LOCA or HELB.

Part II – Buckling Behavior of Steel Containments

Part II of Issue B-5 was also identified in NUREG-0471 and addressed the lack of a well-defined approach for design evaluation of steel containment vessels subject to asymmetrical dynamic loadings that may be limited by the instability of the shell. An acceptable resolution to this issue is to address adequately the design loads, the asymmetrical vessel

configurations associated with the presence of equipment hatches, and the factor of safety in determining allowable loadings.

With respect to Part I of this generic issue, Westinghouse stated in DCD Tier 2 Section 3.8.4.3.1.4 that pressure and thermal loads within or across a compartment (such as main steam isolation valve and SG blowdown compartments) are generated on the basis of postulated HELB. The DCD also stated that, for structural elements including compartment walls and floor slabs, the analysis and design of concrete elements (reinforced concrete structural elements and steel structural modules) conform to American Concrete Institute (ACI) ACI-349 code. The use of ACI-349 code, which provides design criteria and design procedures for the design of reinforced concrete walls and floor slabs under bending and biaxial tension, is acceptable to the staff as discussed in Section 3.8.4 of this report. On this basis, the staff concludes that the concern of Issue B-5, Part I is resolved.

As for Part II of this generic issue, DCD Tier 2 Section 3.8.2.4.1.1 states that the buckling evaluation under external pressure uses the criteria in Article NE-3133 of Section III of the ASME Code. The potential buckling under overall seismic loads are evaluated in accordance with ASME Code, Case N-284, Revision 1. The staff's evaluation and review conclusions for the containment shell buckling under various loads and combined load conditions are discussed in Section 3.8.2 of this report, and include two Open Items (Open Items 3.8.2-1, and 3.8.2-2). On the basis of the discussion above, the concern of Issue B-5, Part II will be resolved upon the resolution of Open Items 3.8.2-1, and 3.8.2-2.

Therefore, based on review of this information, the staff concludes that Task Action Plan Issue B-5, Part II remains unresolved for AP1000 pending the applicant's response to Open Items 3.8.2-1, and 3.8.2-2.

Issue B-17: Criteria for Safety-Related Operator Actions

As discussed in NUREG-0933, Issue B-17 involves the development of a time criterion for safety-related operator actions (SROAs), including a determination of whether automatic actuation is required. This issue also concerns PWR designs that require manual operations to accomplish the switchover from the injection mode to the recirculation mode following a LOCA. Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. Consequently, it becomes necessary to develop appropriate criteria for SROAs. The criteria would include a determination of actions that should be automated in lieu of operator actions and development of a time criterion for SROAs.

The review criteria for this issue are contained in ANSI/ANS 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions." Plants should perform task analysis, simulator studies, and analysis and evaluation of operational data to assess engineered safety features (ESFs) and safety-related control system designs for conformance to the criteria. Where nonconformance is identified, modification of the design and hardware may be required. In DCD Tier 2 Section 1.9.4.2.2, the applicant states that, for the AP1000 design, the safety-related actions required to protect the plant during design-basis events are automatically initiated. The plant systems are designed to provide the required information to

the operator so that plant conditions can be monitored and the performance of the safety-related passive systems and the non-safety-related active systems can be evaluated. The non-safety-related active systems are stated to be designed to automatically actuate, provide defense-in-depth for plant events, and preclude unnecessary actuation of the safety-related passive systems. There is stated to be a backup manual initiation for both the passive and active systems.

The applicant further states that, as described in DCD Tier 2 Chapter 15, the safety systems maintain the plant in a safe condition following design-basis events. This is discussed above in Issue A-31. For most design-basis events, this is accomplished without operator action for up to 72 hours. Operator action is stated to be planned and expected during plant events to achieve the most effective plant response consistent with the event conditions and equipment availability. For events where operator action is taken, the plant design maximizes the time available for operators to complete required actions. For example, the applicant states that, during a SG tube rupture, no operator action is required to establish safe-shutdown conditions or prevent SG overfill. As indicated in Section 18.3, “Element 2: Operating Experience Review,” of this report, the applicant, in WCAP-14645, “Human Factors Engineering Operating Experience Review Report for the AP600 Nuclear Power Plant,” Revision 2, has satisfactorily addressed this item. The Applicant has demonstrated that WCAP-14645 is applicable to AP1000 design, and staff has agreed. Therefore, Issue B-17 is resolved for the AP1000 design.

Issue B-22: LWR Fuel

As discussed in NUREG-0933, Issue B-22 addressed the staff concerns that individual reactor fuel rods sometimes failed during normal operations and many fuel rods are expected to fail during severe core accidents. Failure of fuel rods results in radioactive releases within a plant and is a potential source of release to the public. The resolution of this issue was to ensure that these fuel failures did not result in unacceptable releases to the public. Several problems were identified in the staff analysis to improve the predictability of fuel performance and these were addressed in the revision to SRP Section 4.2, "Fuel System Design," in 1981. The staff concluded that the then-existing requirements on fuel were adequate to ensure continued low fuel defect rates and additional requirements would not significantly increase the number of fuel defects. This issue was then dropped from further consideration.

In DCD Subsection 1.9.4.2.2, the applicant states that the AP1000 reactor core design complies with SRP Section 4.2 and the discussion on the fuel system design is in DCD Tier 2 Section 4.2.

The staff completed its review of the AP1000 fuel assembly design described in DCD Subsection 4.2, which is similar to the 17x17 robust and 17x17 XL robust fuel assemblies. The details of fuel design and acceptance criteria are discussed in Section 4.2 of this report. The staff concludes that this issue resolved.

Issue B-29: Effectiveness of Ultimate Heat Sinks

The applicant identified in DCD Tier 2 Table 1.9-2 that it considered Issue B-29 relevant to the AP1000 design; however, this issue is not required for the AP1000 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-29 addressed the staff concerns identified in NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," June 1978, that the validity of the mathematical models used to predict the performance of dedicated ponds, spray ponds, and cooling towers had not been confirmed, and that better guidance was needed regarding the criteria for the selection of weather data to define the design-basis meteorology. The vulnerability and need for further improvement to the design and operation of ultimate heat sinks (UHS) are addressed in Section 20.3 of this report in Issues 51, 130, and 153. This issue regarded confirming the validity of the NRC mathematical models for prediction of UHS performance and providing guidance regarding the criteria for weather record selection to define UHS design-basis meteorology. This issue was resolved by studies completed by the staff, which confirmed the capabilities of NRC models and provided assurance that the existing guidance was adequate. No new requirements were issued. However, the adequacy of the models to simulate the performance of a plant-specific UHS must be justified on a case-by-case basis.

In DCD Tier 2 Section 1.9.4.2, the applicant states that the passive containment cooling system for the AP1000 design complies with SRP Section 9.2.5, "Ultimate Heat Sink," by providing passive decay heat removal that transfers heat to the atmosphere, which is the UHS for accident conditions. The passive containment cooling system is described in DCD Tier 2 Section 6.2.2.

The staff evaluated the conformance of the AP1000 design to Section 9.2.5 of the SRP in Section 6.2.2 of this report and, on the basis of the staff's conclusions in this section, Issue B-29 is resolved for the AP1000 design.

Issue B-32: Ice Effects on Safety-Related Water Supplies

In DCD Tier 2 Table 1.9-2, the applicant identifies that it considers Issue B-32 relevant to the AP1000 design; however, this issue is not required for the AP1000 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-32 addressed the staff concerns identified in NUREG-0471 that additional information was needed on the potential effects of extreme cold weather and ice buildup on the reliability of plant water supplies. Experience gained during past severe winters indicated that a more thorough understanding of the potential effects of severe ice conditions was necessary to confirm that the design and operation of safety-related water supplies would ensure adequate operation of safety systems. Guidance for the review of licensee submittals regarding ice effects is in SRP Section 2.4.7, "Ice Effects."

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that DCD Tier 2 Section 6.2.2 describes the UHS design and discusses the features that prevent freezing in the passive containment cooling system. This issue was addressed and resolved through the resolution of Issue 153, which is discussed for the AP1000 design in Section 20.3 of this report. Therefore, on the basis of the staff's conclusions in this section, Issue B-32 is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue B-36 addressed the staff concern that the then-current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units needed to be revised. This issue was resolved by the issuance of Revision 3 of RG 1.52 for ESF ventilation filter units in June 2001, and Revision 2 of RG 1.140 for normal atmosphere cleanup systems June 2001.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that there are no safety-related air filtration systems in the AP1000 design. The specific functions of the normal ventilation systems are outlined in DCD Tier 2 Sections 6.4 and 9.4.1, with a discussion on the conformance with RG 1.140 in DCD Tier 2 Appendix 1A.

The staff determined that Issue B-36 is closed for the AP1000 design because the non-radioactive ventilation system (VBS) and the containment air filtration system (VFS) conform to RG 1.140. For the defense-in-depth filtration function of the VBS and VFS, DCD Tier 2 Appendix 1A provides a comparison of the AP1000 design to RG 1.140. In addition, DCD Tier 2 Section 9.4 provides direct reference to DCD Tier 2 Appendix 1A. Therefore, Issue B-36 is resolved for the AP1000 design.

Issue B-53: Load Break Switch

GDC 17 of Appendix A to 10 CFR Part 50 requires that two offsite circuits be available to supply vital plant loads following a loss of all onsite ac power supplies. For those plants with designs that rely on a generator load break switch (or circuit breaker), the switch (or breaker) is relied on to isolate the main generator from the main transformer following a turbine trip to allow power to be fed from the grid through the main transformer as a second offsite power source to the onsite Class 1E power system.

The AP1000 design incorporates a generator load circuit breaker to provide a reliable source of ac power to the electrical systems; however, the AP1000 design does not require ac power sources for design-basis accidents.

Therefore, Issue B-53 is not applicable for the AP1000 design.

Issue B-56: Diesel Reliability

Issues that result in a loss of offsite power necessitate reliance on the onsite emergency diesel generators for successful accident mitigation. Improvement of the starting reliability of onsite emergency diesel generators would reduce the probability of events that could lead to core-melt accident.

The AP1000 diesel generators are non-Class 1E and are not required for accident mitigation, and their reliability is founded on industry standards and practices.

Therefore, Issue B-56 is not applicable and is considered resolved for the AP1000 design.

Issue B-60: Loose Parts Monitoring System

The presence of a loose object in the primary coolant system can be indicative of degraded reactor safety resulting from failure or deterioration of a safety-related component. As discussed in NUREG-0933, Issue B-60 addressed the need to have a loose part detection program for early detection of loose metallic parts in the primary system. The NRC has developed loose-parts detection system hardware criteria and programmatic criteria for loose-parts detection programs, as described in RG 1.133, Revision 1. All construction permits and operating licenses reviewed after January 1, 1978, were required to meet the provisions of RG 1.133, Revision 1. Thus, this issue was resolved and no new requirements were established.

In DCD Tier 2 Table 1.9-2, the applicant indicates that Issue B-60 regarding loose-part detection system was resolved with no new requirements. As described in DCD Subsection 4.4.6.4, the AP1000 design has a digital metal impact monitoring system (DMIMS), which conforms with RG 1.133., for monitoring the RCS for metallic loose parts.   The staff evaluation of the AP1000 DMIMS is discussed in Section 4.4.4.2 of this report. Therefore, the staff concludes Issue B-60 is resolved for AP1000.

Issue B-61: Allowable ECCS Equipment Outage Periods

As discussed in NUREG-0933, Issue B-61 addresses establishing surveillance test intervals and allowable equipment outage periods, using analytically based criteria and methods for the TSs. The present TS-allowable equipment outage intervals and test intervals were determined primarily on the basis of engineering judgment. Studies performed by the NRC on operating reactors indicated that from 30 to 80 percent of the emergency core cooling system (ECCS) unavailability was the result of testing, maintenance, and allowed outage periods. Therefore, by optimizing the allowed outage period and the test and maintenance interval, the equipment unavailability and public risk can be reduced.

In DCD Tier 2 Section 1.9.4.2.2, Westinghouse states that 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 TSs, a combination of NUREG-1431 precedent, system design, and safety-related function is considered.

The staff's evaluation of the AP1000 TSs is in Section 16 of this report. On the basis of this evaluation and Westinghouse's response to this issue, Issue B-61 is resolved for the AP1000 design.

Issue B-63: Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary (RCPB)

Issue B-63 addresses the adequacy of the isolation of low-pressure systems that are connected to the RCPB. Several systems connected to the RCPB in operating plants have design pressures that are considerably below the RCS operating pressure. The NRC has recommended that valves forming the interface between these high- and low-pressure systems associated with the RCPB have sufficient redundancy to ensure that the low-pressure systems are not subjected to pressures beyond their design limits.

The resolution of this issue for the AP1000 has been subsumed by the resolution of GSI 105, which is discussed in Section 20.3 of this report. Therefore, Issue B-63 is resolved for the AP1000 design.

Issue B-66: Control Room Infiltration Measurements

The control room area ventilation systems and control building layout and structures are reviewed to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases, and that the control room can be maintained as the backup center from which technical personnel can safely operate during an accident. A key parameter affecting control room habitability is the rate of air infiltration into the control room. Current estimates of these rates are dependent on data relating to buildings that are substantially different from typical control room buildings in nuclear power plants.

As discussed in NUREG-0933, Issue B-66 was intended to facilitate compliance with staff requirements and guidance on control room habitability, specifically (1) GDC 19 and (2) SRP Sections 6.4, "Control Room Habitability System," and 9.4.1, "Control Room Area Ventilation System." Additional experimentally measured air exchange rates of operating reactor control rooms resulted in Revision 2 of SRP Section 6.4. See also the resolution of Issues 83 and III.D.3.4 for the AP1000 design in Sections 20.3 and 20.4, respectively, of this report.

In DCD Tier 2 Section 1.9.4.2.2, the applicant states that the Main Control Room (MCR) for the AP1000 design is essentially leak-tight. Unfiltered air in-leakage is minimized by maintaining the MCR at a slightly positive pressure and the verification of the design infiltration rate is in accordance with SRP Section 6.4. Control room habitability is discussed in DCD Tier 2 Section 6.4.

In DCD Tier 2 Sections 6.4.5.1 and 14.2.9.1.6, the applicant committed to performing preoperational testing for in-leakage during main control room emergency habitability system (VES) operation in accordance with American Society for Testing Materials (ASTM) E741-2000, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." In addition, in DCD Tier 2 6.4.5.4, the applicant committed to conducting testing for

MCR in-leakage during VES operation in accordance with ASTM E741-2000. the applicant also committed to revise DCD Tier 2 Section 6.4.7 to state that COL applicant will provide the testing frequency for the main control room inleakage test. Issue B-66 is resolved because the staff concluded that the testing described above will ensure that the AP1000 design meets the dose limits of GDC 19.

Issue C-1: Hermetic Seals on Instrumentation and Electrical Equipment

Unresolved Safety Issue A-24 addresses the equipment qualification (EQ) of safety-related instrumentation and electrical equipment that may be required to function under accident conditions. This program also confirms the integrity of seals employed in the design of Class 1E equipment. Therefore, Issue C-1 is resolved for the AP1000 design.

Issue C-4: Statistical Methods for ECCS Analysis

As discussed in NUREG-0933, Issue C-4 addressed the statistical methods used for performance evaluation of the ECCS during a LOCA. In accordance with the requirements of 10 CFR 50.46, "Acceptance Criteria for ECCS for Light-Water Nuclear Power Reactors," as amended on September 16, 1988, the NRC requires that the LOCA analyses for license applications use either the 10 CFR Part 50, Appendix K, evaluation models, or the realistic models which statistically accounts for uncertainties, including the uncertainty of calculation in the adverse direction. The realistic models must be supported by applicable experimental data. Uncertainties in the realistic models and input must be identified and assessed so that uncertainty in the calculated results can be estimated.

Appendix K of 10 CFR Part 50 specifies the requirements for LWR ECCS analysis, which call for specific conservatism to be applied to certain models and correlations used in the analysis to account for data uncertainties at the time Appendix K was written. USI C-4 addressed NRC development of a statistical assessment of the uncertainty level of the peak cladding temperature limit. In 1988, 10 CFR 50.46 was revised to allow the realistic ECCS evaluation model, in addition to the evaluation model conforming to the Appendix K requirements. This best estimate evaluation model will use an analytical technique that realistically describes the behavior of the reactor system during a LOCA, with comparisons to applicable experimental data. The realistic evaluation model must identify and account for uncertainties in the analysis method and inputs so that when the calculated ECCS cooling performance is compared to the acceptance criteria, there is a high level of probability that the criteria would not be exceeded.

In DCD Subsection 1.9.4.2.2, the applicant states that the AP1000 methodology applied for LOCA analysis is discussed in DCD Tier 2 Chapter 15.

As described in DCD Tier 2 Chapter 15, the computer codes WCOBRA/TRAC and NOTRUMP, respectively, are used for the large- and small-break LOCA analyses. WCOBRA/TRAC is a realistic code, and the uncertainties will be included in the analysis. NOTRUMP is a code using the Appendix K requirements. The staff provides its evaluation of the acceptability of these codes for the AP1000 application in Section 21 of this report. Therefore, Issue C-4 is resolved for the AP1000 design.

Issue C-5: Decay Heat Update

As discussed in NUREG-0933, Issue C-5, addressed the specific decay heat models for the LOCA analysis models. This issue involved following the work of research groups in determining best-estimate decay heat data and associated uncertainties for use in LOCA calculations.

In accordance with the requirements of 10 CFR 50.46, as amended on September 16, 1988, the LOCA analyses for license applications should use either the 10 CFR Part 50, Appendix K, models, or the realistic models supported by applicable experimental data and including uncertainty of calculation in the adverse direction. When Appendix K models are used, Appendix K requires the use of 1971 ANS Standard, ANS-5, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," times 1.2, for the heat generation rates from the radioactive decay of fission products in the ECCS calculation. When realistic models are used, the staff has determined that the 1979 ANSI/ANS Standard 5.1, "Decay Heat Power in Light-Water Reactors," is technically acceptable for licensing applications.

In DCD Subsection 1.9.4.2.2, the applicant states that the large-break LOCA analyses for the AP1000 design, discussed in DCD Subsection 15.6.5, use the decay heat model identified in the 1979 ANSI 5.1 standard.

For the AP1000 application, the 1971 ANS decay heat model and the 1979 ANSI/ANS decay heat model are used in NOTRUMP and WCOBRA/TRAC, respectively, for small- and large-break LOCAs. The staff has completed and documented its review of small- and large-break LOCA analyses using NOTRUMP and WCOBRA/TRAC, respectively, in Chapter 15 of this report. The staff considers Issue C-5 resolved for the AP1000 design.

Issue C-6: LOCA Heat Sources

As discussed in NUREG-0933, Issue C-6 addressed the issue identified in NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D),” dated June 1978, that involved staff evaluations of vendors' data and approaches for determining LOCA heat sources and the need for developing staff positions. The contributors to LOCA heat sources, along with their associated uncertainties and the manner in which they are combined, have an impact on LOCA calculations. The staff informed the Commission in SECY-83-472, "Emergency Core Cooling System Analysis Methods," November 17, 1983, that statistical combination of LOCA heat sources would be allowed to justify the relaxation of non-required conservatism in ECCS evaluation models.

In DCD Tier 2 Subsection 1.9.4.2.2, the applicant states that the discussion of LOCA heat sources for the AP1000 design is included in DCD Subsection 15.6.5. The staff completed and documented its review of small- and large break LOCA analyses using NOTRUMP and WCOBRA/TRAC, respectively, in Chapter 15 of this report. The staff considers Issue C-6 resolved for the AP1000 design.

Issue C-10: Effective Operation of Containment Sprays in a LOCA

As discussed in NUREG-0933, Issue C-10 addressed the effectiveness of various containment sprays to remove airborne radioactive material that could be present within the containment following a LOCA. This was expanded to include the possible damage to equipment located within the containment as a result of an inadvertent actuation of the sprays.

The AP1000 relies on natural mechanisms, which are enhanced by the Passive Containment System (PCS), for the removal of airborne radioactive material post-LOCA. The staff's evaluation of these natural removal mechanisms (such as holdup, sedimentation, and diffusion) can be found in Section 15.3 of this report. In a Staff Requirements Memorandum, dated June 30, 1997, the Commission approved the staff's recommendation that the AP1000 include a containment spray system or equivalent for accident management following a severe accident. The containment spray system is described in DCD Tier 2 Section 6.5.2 and the staff's evaluation of the system is in Section 19.2.3.3.9, "Non-Safety-Related Containment Spray System," of this report. The applicant concluded in DCD Tier 2 Section 6.5.2 that inadvertent actuation of the containment spray system was not credible. The staff's evaluation of this conclusion is in Section 6.2.1.1 of this report. On the basis of the staff's evaluations in Sections 6.2.1.1 and 15.3 of this report, Issue C-10 is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue C-17 was intended to develop criteria for the acceptability of radwaste solidification agents to properly implement a process control program for packaging diverse radioactive plant wastes for shallow land burial. There are no current criteria for a finding of acceptability of solidification agents.

As stated in NUREG-0933, the Commission issued 10 CFR Part 61 on licensing requirements for land disposal of radioactive waste, including Section 61.56, which addresses acceptable waste characteristics. Also, the staff developed BTP ETSB 11-3 to be part of SRP Section 11.4, "Solid Waste Management Systems," and provide design guidance for solid waste management systems (SWMSs) to be used at LWRs. Therefore, this issue has been resolved for implementation at nuclear power plants.

In DCD Tier 2 Section 1.9.4, the applicant states that the solid radwaste system for the AP1000 design transfers, stores, and prepares spent ion exchange resins for disposal. The system 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. This will be provided using mobile systems. Solidification of waste is not performed by permanently installed systems.

The staff evaluated the conformance of the AP1000 design to Section 11.4 of the SRP in Section 11.4 of this report. On the basis of the staff's conclusions in this section, Issue C-17 is resolved for the AP1000 design.

20.3 New Generic Issues

The new generic issues of NUREG-0933 listed in Table 20.1-1 are evaluated against the AP1000 design in this section. The majority of the items were chosen either because (1) 10 CFR 52.47(a)(1)(iv) or 10 CFR 52.34(f) require the design to comply with them, or (2) the applicant decided that the item applied to the design and included a discussion of the item in DCD Tier 2.

Issue 14: PWR Pipe Cracks

As discussed in NUREG-0933, Issue 14 addressed cracking in PWR non-primary (i.e., secondary) piping systems as a result of stress corrosion, vibratory and thermal fatigue, and dynamic loading. Cracking in PWR non-primary system piping could lead to a decrease of the system functional capability and could possibly result in such situations as degraded core cooling. This issue deals with occurrences of main feed water (MFW) line cracking in certain Westinghouse and Combustion Engineering PWRs. In September 1980, the PWR Pipe Study Group completed its investigation of the issue and published its findings in NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping of Pressurized Water Reactors." This report provided conclusions regarding systems safety and recommended technical solutions to the issue.

The staff developed recommendations that included augmented inspections requirements, but concluded that they had low risk-reduction value. Therefore, this issue was resolved and no new requirements were established. Other recommendations by the staff included upgrading ASME Section V and Section XI ultrasonic testing (UT) procedures and requirements to achieve more reliable flaw detection and characterization. Upgrades to ASME Section V and Section XI have occurred progressively since 1980, and include the development of the ASME Code Section XI, Appendix VIII, supplements incorporated by reference in 10 CFR 50.55a. These requirements have resulted in more reliable flaw detection and characterization through performance demonstration requirements on equipment, personnel, and procedures.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the design and inspection requirements for feedwater lines are in DCD Tier 2 Section 10.4.7. Further, the issue of ISI of Class 2 and 3 components is addressed in DCD Tier 2 Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” which is evaluated in Section 6.6 of this report. Section 6.6 discusses weld accessibility for inspection purposes and compliance with ASME Code inspection requirements. On this basis, Issue 14 is resolved for the AP1000 design.

Issue 15: Radiation Effects on Reactor Vessel Supports

As discussed in NUREG-0933, Issue 15 addresses the potential for radiation embrittlement of reactor vessel support structures. Neutron irradiation of structural materials causes embrittlement that may increase the potential for propagation of flaws that might exist in the materials. The potential for brittle fracture of these materials is typically measured in terms of the material's nil ductility transition temperature (NDTT). As long as the operating environment in which the materials are used has a higher temperature than the material's NDTT, failure by

brittle fracture is not expected. Many materials, when subjected to neutron irradiation, experience an upward shift in the NDTT, that is, they become more susceptible to brittle fracture at the operating temperatures of interest. This effect has to be accounted for in the design and fabrication of reactor vessel support structures.

As discussed in NUREG-0933, this issue had a high-priority ranking; but after extensive evaluation, the staff concluded that no new requirements needed to be issued by the NRC.

In DCD Tier 2 Section 1.9.4.2.3, applicant states that the supports for the AP1000 reactor vessel are designed for loading conditions and environmental factors, including the neutron fluence. The material requirements are stated to include fracture toughness requirements and impact testing requirements in compliance with ASME Code, Section III, Subsection NF. These supports are not in the region of high neutron fluence where neutron radiation embrittlement of the supports would be a significant concern.

On the basis of the above, the staff considers the reactor vessel supports for the AP1000 design to be adequately designed for radiation effects, and Issue 15 is resolved for the AP1000 design.

Issue 22: Inadvertent Boron Dilution Events

As discussed in NUREG-0933, Issue 22 addressed the possibility of core criticality during cold-shutdown conditions from inadvertent boron dilution events. Although this issue was resolved with no new requirements, the acceptance criterion is that plants shall minimize the consequences of such events by meeting SRP Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)." Specifically, the plant shall respond in such a way that the criteria regarding fuel damage and system pressure are met, and the dilution transient is terminated before the shutdown margin is eliminated. If operator action is required to terminate the transient, redundant alarms must be in place and the following minimum time intervals must be available between an alarm announcing an unplanned dilution and when shutdown margin is lost:

- 30 minutes during refueling (Mode 6)
- 15 minutes during all other operating modes

In DCD Tier 2 Section 15.4.6, the applicant provides a safety analysis for AP1000 that demonstrates that redundant alarms are available to enable operators to detect and terminate an inadvertent boron dilution event within the above required time intervals, before shutdown margin is lost.

In addition to the events in this issue, the staff identified the following two boron dilution scenarios where a deborated water slug may accumulate in the Reactor Coolant System (RCS) and a restart of the Reactor Coolant Pumps (RCPs) will cause this slug to pass through the core, resulting in criticality or a power excursion:

- The first scenario occurs during a plant startup when the reactor is deborated as part of startup procedures. A loss of offsite power will result in tripping the RCPs and charging pump. The subsequent startup of the diesel generator will restart the charging pump and cause the accumulation of deborated water in the reactor lower plenum. The RCP restart with recovery of offsite power will cause this deborated water to pass through the core.

- The second scenario is related to transients or accidents, such as a small-break LOCA with heat removal by reflux condensation natural circulation that may result in an accumulation of deborated water in the RCS loop. This water will pass through the core with an inadvertent restart of the RCPs.

The staff completed and documented its review of inadvertent boron dilution issues in Sections 15.2.4.6 and 15.2.8 of this report. The staff considers Issue 22 resolved for the AP1000 design.

Issue 23: Reactor Coolant Pump (RCP) Seal Failures

As discussed in NUREG-0933, Issue 23 addressed the concerns about RCP seal failures that could cause a small-break LOCA. PRA analyses have indicated that the overall probability of core damage as a result of a small-break could be dominated by RCP seal failures. This issue includes improving the reliability of RCP seals by reducing the probability of seal failure during normal operations and under abnormal conditions. Specifically, acceptable resolutions to this issue include an RCP seal design that ensures the RCP seal integrity following SBO for an extended period.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the AP1000 RCPs are canned motor pumps that contain the motor and all rotating components inside a pressure vessel designed for full RCS pressure. The applicant states that the shaft for the pump impeller and rotor is within this vessel; therefore, seals are not required. Further discussion on the canned motor pump design is in DCD Tier 2 Section 5.4.1. The applicant concludes that because the RCPs do not rely on seals as being part of the RCPB, Issue 23 is not applicable to the AP1000 design.

The staff agrees that the AP1000 design uses canned motor RCPs, which contain the motor and all rotating components inside a pressure vessel designed for full RCS pressure. The shaft for the impeller and rotor is contained within the pressure boundary; therefore, the staff concludes that seals are not required to restrict leakage out of the pump into containment, and Issue 23 does not apply to the AP1000 design. On the basis of the above, Issue 23 is resolved for the AP1000 design.

Issue 24: Automatic ECCS Switchover To Recirculation

Issue 24 addresses the staff's concerns following a review of operating events that indicated a significant number of ECCS spurious actuations, particularly the four events that occurred at the Davis-Besse plant during 1980. Switchover from injection to recirculation involves realignment of several valves, and may be achieved by (1) manual realignment, (2) automatic

realignment, or (3) a combination of both. Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures. The safety significance of the issue is that switching suction to the sump prematurely could adversely affect the accident because the containment sump may not have enough inventory to provide pump suction. In NUREG-0933, this issue was classified as medium-safety priority, but had not been generically resolved.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the AP1000 does not switch from injection to recirculation in the sense that injection is not isolated when recirculation is opened, and that the AP1000 does provide for automatic opening of the recirculation line on a low level signal from the in-containment refueling water storage tank (IRWST). The staff notes that the AP1000 passive safety system design does not have safety-related pumps, as do the plants originally addressed by Issue 24. Furthermore, if the recirculation line were opened in the AP1000, the flow path from the IRWST to the reactor vessel would still exist. This is different from conventional PWRs where the flow path from the refueling water storage tank would be closed when recirculation mode is entered. Therefore, for the AP1000, the situation is not analogous to that addressed by Issue 24 for operating PWRs, and is not applicable to AP1000 design.

Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

As discussed in NUREG-0933, Issue 29 addressed staff concerns about the number of events involving the degradation of threaded fasteners (such as bolt cracking, corrosion and failure) in operating plants from 1964 to the early 1980s. Many of the events were related to components of the RCPB and support structures of major components. This raised questions about the integrity of the RCPB and the reliability of the component support structures following a LOCA or a seismic event. The licensees reported failures involving a variety of threaded fasteners and most frequently reported degradation mechanisms were wastage (corrosion) from boric acid attack and stress corrosion cracking (SCC). The former occurred more often at RCPB joints; the latter in structural bolting.

This issue was resolved and no new requirements were established on the basis of (1) operating experience with bolting in both nuclear and conventional power plants; (2) actions already taken through bulletins, generic letters, and information notices since 1982; and (3) industry-proposed recommendations and actions, which are documented in the EPRI Reports NP-5769 ("Degradation and Failure of Bolting in Nuclear Power Plants," April 1988) and NP-5067 ("Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel," Volume 1: "Large Bolt Manual," 1987 and Volume 2: "Small Bolts and Threaded Fasteners," 1990). The resolution of this issue is documented in GL 91-17, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants," dated October 17, 1991; and NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," dated June 30, 1990.

In DCD Tier 2 Section 1.9.4.2.3, applicant states that the elements of resolution of this issue pertain to operational and maintenance practices, which will be addressed by the COL applicant. It also states that conformance to the ASME Code, Section III requirements for

pressure boundary components and related supports, which the AP1000 design meets, will provide safe operation in the event of bolting degradation. Further, because of the emphasis in the AP1000 design on access for maintenance and inspection, the recommended maintenance practices can be readily implemented.

The staff concludes that applicant has adequately addressed this issue for the AP1000 design; therefore, Issue 29 is resolved for the AP1000 design.

Issue 43: Reliability of Air Systems

As discussed in NUREG-0933, Issue 43 is not required for the AP1000 design to meet 10 CFR 52.47(a)(1)(ii) or (iv); however, the staff believed it should be addressed for the AP1000 design because the issue dealt with all causes of air system unavailability. The issue addressed the incident at Rancho Seco where desiccant particles in the valve operator caused the slow closure of a containment isolation valve. Desiccant contamination in the instrument air system (IAS) was also found to be a contributing cause of the loss of the salt water cooling system at San Onofre in March 1980; this incident resulted in Issue 44, "Failure of the Saltwater Cooling System." Because the only new generic concern found in the evaluation of the San Onofre event was the common-cause failure of safety-related components as a result of contamination of the IAS, Issue 44 was combined with Issue 43.

Issue 43 was broadened to include all causes of air system unavailability because U.S. LWRs rely upon air systems to actuate or control safety-related equipment during normal operation even though they are not safety-grade systems at most operating plants. Safety system design criteria require (and plant accident analyses assume) that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform its intended function with the assistance of backup accumulators. An NRC Office for Analysis and Evaluation of Operational Data (AEOD) case study highlighted 29 failures of safety-related systems that resulted from degraded or malfunctioning air systems. These failures contradict the requirement that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or will perform its intended function with the assistance of backup accumulators. Some of the systems that may be significantly degraded or failed are decay heat removal, auxiliary feedwater, boiling-water reactor scram, main steam isolation, salt water cooling, emergency diesel generator, containment isolation, and the fuel pool seal system. The end result of degradation or failure of safety or safety-related systems is an increase in the expected frequency of core-melt events and, therefore, an increase in public risk.

This issue was resolved by the issuance of GL 88-14, "Instrument Air Supply Problems Affecting Safety-Related Equipment," dated August 8, 1988, which required licensees and applicants to review the recommendations of NUREG-1275 ("Operating Experience Feedback Report – Air Systems Problems," two volumes, dated July and December 1987, respectively) and perform a design and operations verification of the IAS. The following is a discussion of

the purposes for which the applicant considered the recommendations in NUREG-1275, Volume 2, for the AP1000 design:

- Ensure that air system quality is consistent with equipment specifications and is periodically monitored and tested.

  In DCD Tier 2 Section 9.3.1, the applicant states that in accordance with NUREG-1275, instrument air quality meets the manufacturer’s standards for pneumatic equipment supplied as part of the plant. In addition, periodic checks are made to assure high-quality instrument air as specified in ANSI/ISA-S7.3, “Quality Standard for Instrument Air.”

- Ensure adequate operator response by formulating and implementing anticipated transient and system recovery procedures for loss-of-air events.

  In DCD Tier 2 Section 9.3.7, the applicant states that the COL applicant will address DCD Tier 2 1.9.4.2.3, Issue 43 as part of training and procedures identified in DCD Tier 2 Section 13.5.

- Improve training to ensure that plant operations and maintenance personnel are sensitized to the importance of air systems to common mode failures.

  In DCD Tier 2 Section 9.3.7, the applicant states that the COL applicant will address DCD Tier 2 1.9.4.2.3, Issue 43, as part of training and procedures identified in DCD Tier 2 Section 13.5.

- Confirm the adequacy and reliability of safety-related backup accumulators.

  In DCD Tier 2 Section 9.3.1, the applicant states that there are no safety-related air operated valves that rely on safety-related air accumulators to actuate to the fail safe position upon loss of air pressure.

- Verify equipment response to gradual losses of air to ensure that such losses do not result in events that fall outside final safety analysis report (FSAR) analysis.

  In DCD Tier 2 Section 9.3.1.4, the applicant states that during initial plant testing before reactor startup, safety systems utilizing instrument air will be tested as part of the safety system test to verify fail-safe operation of air-operated valves upon sudden loss of instrument air or gradual reduction of air pressure as described in RG 1.68.3, “Preoperational Testing of Instrument and Control Air Systems.”

The items above are adequately addressed for the AP1000. Therefore, the staff finds Issue 43 resolved for the AP1000 design.

Issue 45: Inoperability of Instruments Due to Extreme Cold Weather

As discussed in NUREG-0933, Issue 45 addressed the potential for safety-related equipment instrument lines to become inoperable as a result of freezing or reaching the precipitation point of the sensing fluids. Typical safety-related systems employ pressure and level sensors that use small-bore instrumentation lines. Most operating plants contain safety-related equipment and systems, parts of which are exposed to ambient temperature conditions. These lines generally contain liquid (e.g., borated water) that is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to subfreezing temperatures, heat tracing or insulation or both is used to minimize the effects of cold temperatures. These sensing lines are of concern because, should they freeze, they may prevent a safety-related system or component from performing its safety function.

To resolve this issue, the staff issued RG 1.151, "Instrument Sensing Lines," to supplement the existing guidance and requirements in the SRP, applicable GDC, and Instrument Society of America (ISA) standard ISA-67.02, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants." RG 1.151 addresses the prevention of freezing in safety-related instrument-sensing lines and includes such design issues as diversity, independence, monitoring, and alarms. In February 1984, SRP Sections 7.1, "Instrumentation and Controls – Introduction," Revision 3; Appendix A, Revision 1, to Section 7.1; and 7.7, "Control Systems," Revision 3 were revised to incorporate the resolution of this issue. Thus, this issue was resolved and new requirements were issued.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the AP1000 design complies with SRP Sections 7.1; Appendix A to Section 7.1; Section 7.5, "Information Systems Important to Safety"; and Section 7.7, “Control Systems.” The conformance of the AP1000 design to RG 1.151 is addressed in DCD Tier 2 Appendix 1A.

On the basis of this, the staff concludes that the AP1000 design complies with the relevant sections of RG 1.151 and the updated SRP sections. Therefore, Issue 45 is resolved for the AP1000 design.

Issue 51: Improving the Reliability of Open-Cycle Service Water Systems

As discussed in NUREG-0933, Issue 51 addressed fouling of safety-related open-cycle service water systems by either mud, silt, corrosion products, or aquatic bivalves. This problem has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation in nuclear power plants. This issue was originally to address only aquatic bivalves. However, the issues on flow blockage in essential equipment caused by Corbicula (Issue 32) and service water system flow blockage caused by Blue Mussels (Issue 52) were incorporated into this issue, and Issue 51 was expanded to consider if the NRC staff should develop new requirements for improving the reliability of open cycle water systems. New requirements were issued in GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989, on baseline fouling programs for nuclear power plants.

In DCD Tier 2 Section 1.9.4, the applicant states that the service water system for the AP1000 design provides cooling water to the component cooling water system and has no safety-related functions. It is stated that none of the safety-related equipment requires water cooling to effect a safe shutdown or mitigate the effects of design-basis events. Heat transfer to the Ultimate Heat Sink (UHS) 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-term corrosion and organic fouling are discussed in DCD Tier 2 Section 9.2.1.

On the basis of the staff's review, which is discussed in Section 9.2.1 of this report, the staff concludes that the service water system is adequately designed to minimize fouling, and Issue 51 is resolved for the AP1000 design.

Issue 57: Effects of Fire-Protection Systems Actuation on Safety-Related Equipment

NUREG-0933, "Generic Issues," Issue 57, and NUREG-5580, "Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety Related Equipment," addressed fire protection system (FPS) actuations that have caused adverse interactions with safety-related equipment at operating nuclear power plants. Experience has shown that safety-related equipment subjected to water spray, as from the FPS, could be rendered inoperable and that numerous spurious actuations of the FPS have been initiated by operator testing errors or by maintenance activities, steam, or high humidity in the vicinity of FPS detectors.

DCD Tier 2 Section 9A.3.1.1, "Containment/Shield Building," specifies that inadvertent operation of an automatic suppression system is prevented by the normally closed containment isolation valve in the water supply line. Operator action is required to open this valve and admit water to the system. Therefore, because the AP1000 design does not provide automatic fire suppression in safety-related areas, Issue 57 for the AP1000 design is considered resolved.

Issue 67.3.3: Improved Accident Monitoring

As discussed in NUREG-0933, Issue 67.3.3 addressed weaknesses in reactor system monitoring that could inhibit correct operator responses to events similar to the SG tube rupture (SGTR) event at the Ginna Power Plant on January 25, 1982. During the event, weaknesses in accident monitoring were apparent including (1) non redundant monitoring of RCS pressure, (2) failure of the position indication for the SG relief and safety valves, and (3) limited range of the charging pump flow indicator. As stated in NUREG-0933 and Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980, (Supplement 1, January 1983), the implementation of the recommendations described in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Revision 2, December 1980, resolved this issue.

In DCD Tier 2 Section 1.9.4.2.3, Response to Issue 67.3.3, the applicant stated that the guidance of RG 1.97 is followed to determine the appropriate parameters to monitor in the AP1000 design. The post-accident monitoring system is described in Section 7.5.

The staff concludes, as stated in Section 7.5 of this report, that the post-accident monitoring system conforms to Revision 3 of RG 1.97 and is acceptable. The staff concludes that the Issue 67.3.3 is resolved for the AP1000 design.

Issue 70: Power-Operated Relief Valves (PORV) and Block Valve Reliability

PORVs and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation; the block valves were installed because of expected leakage from the PORVs. Neither valve type was needed to safely shut down a plant or mitigate the consequences of accidents. In 1983, the staff determined that PORVs were relied on to mitigate design-basis SGTR accidents and questioned the acceptability of relying on non-safety-grade components to mitigate design-basis accidents (DBAs). NUREG-0933, Issue 70, addressed the assessment of the need for improving the reliability of PORVs and block valves.

In DCD Tier 2 Section 1.9.3, Item (1)(iv), the applicant states that the AP1000 design does not include PORVs. Overpressure protection is provided by two totally enclosed pop-type safety valves. If the pressurizer pressure exceeds the set pressure, the safety valves lift. A temperature indicator in the discharge piping for each safety valve alarms on high temperature to alert the operator to when the valves open. The staff concludes that because the AP1000 design does not include PORVs and block valves, Issue 70 is not applicable. Therefore, Issue 70 is resolved for the AP1000 design.

Issue 73: Detached Thermal Sleeves

As discussed in NUREG-0933, Issue 73 addressed the staff concerns, during the period 1978 to 1980, about reports of fatigue failures of thermal sleeve assemblies in the piping systems of both PWRs and BWRs. There have been five generations (0 through 4) of thermal sleeves used in the applicant reactors. Only "Generation 3" thermal sleeves have been found to be susceptible to high-cycle stresses due to flow-induced vibrations because of the particular weld attachments used in that design. The vibrations caused fatigue failures at the attachment welds and subsequent cracking and tearing away of the thermal sleeves. This issue was applicable to the design and operation of approximately 20 of the applicant plants that used that generation thermal sleeve. This issue was resolved for the applicant plants with the publication of NUREG/CR-6010, "History and Current Status of Generation 3 Thermal Sleeves in the applicant Nuclear Power Plants," July 1992.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the AP1000 does not use Generation 3 thermal sleeves. Based on the staff review of this information, New Generic Issue 73 is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue 75 addressed the generic implications of two events at Salem Unit 1 where there were failures to scram automatically because of the failure of both reactor trip breakers to open on receipt of an actuation signal. This issue was expanded to include a number of issues raised by the staff that were closely related to the design and testing of the Reactor Protection System (RPS). The requirements for this issue were stated in GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Event," dated July 8, 1983.

The actions covered by GL 83-28 fell into the following four areas:

(1) Post-Trip Review – This action addresses the program, procedures, and data collection capability to ensure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

(2) Equipment classification and vendor interface –This action addresses the programs for ensuring that all components necessary for performing required safety-related functions are properly identified in documents, procedures, and information-handling systems that are used to control safety-related plant activities. In addition, this action addresses the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

(3) Post-maintenance testing – This action addresses post-maintenance operability testing of safety-related components.

(4) RTS reliability improvements – The intent of this action is to ensure that (a) vendor-recommended reactor trip breaker modifications and associated RPS changes are completed in PWRs, (b) a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in PWRs, (c) the shunt trip attachment activates automatically in all PWRs that use circuit breakers in their RTS, and (d) online functional testing of the RTS is performed on all LWRs.

The AP1000 design of the reactor trip breakers and the RPS is outlined in DCD Tier 2 Section 7.1. Information on the functional requirements for reactor trip and conformance with industry and regulatory guidance is outlined in DCD Tier 2 Section 7.2. The provisions provided to display and record parameters used by the reactor trip system are outlined in DCD Tier 2 Sections 7.1.2.6 and 7.1.2.13. DCD Tier 2 Section 7.5 also provides information on requirements for safety-related display information. Based on the staff review of this information, New Generic Issue 75 is resolved for the AP1000 design.

Issue 79: Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

As discussed in NUREG-0933, Issue 79 addressed the concern for an unanalyzed reactor vessel thermal stress during natural convection cooldown (NCC) of PWR reactors. The concern emerged from a preliminary evaluation of the voiding event that occurred in the upper head of the St. Lucie Unit 1 reactor on June 11, 1980. On the basis of several conservative assumptions, B&W tentatively concluded that during natural convection cooling, axial temperature gradients could develop in the vessel flange area, which could produce thermal stresses in the flange area, or in the studs, that might exceed values allowed by the ASME Code, Section III when added to the stresses already considered (such as boltup loads or pressure loads).

The staff's efforts to resolve this issue were based on a review of a B&W NCC analysis and the results of a NCC analysis by a NRC contractor, both of which were performed for the B&W 177 fuel assembly reactor vessel. The staff's evaluation and resolution of Issue 79 is documented in NUREG-1374, "An Evaluation of PWR Reactor Vessel Thermal Stress During NCC," dated May 1991, and GL 92-02. On the basis of conservative analyses and qualitative extrapolation of the results, the staff concluded the following in NUREG-1374:

- The B&W 177 is considered analyzed for NCC events that are bounded by the NCC transient profile shown in Figure 3 of NUREG-1374. The bounding profile in this figure was generated by the staff's contractor by using a conservative assumption of a maximum cooldown rate of 100° F per hour during the NCC event. This profile was used by the contractor in its conservative confirmatory stress analysis of the B&W 177.

- Adequate geometric similarity exists between the B&W 177 and other U.S. PWRs to support extending the findings and conclusions in NUREG-1374 to all U.S. PWRs.

- It is extremely unlikely that a single NCC event will cause the failure of any existing U.S. PWR reactor vessel, even if a cooldown rate of 100° F per hour is exceeded.

- NCC events of the type analyzed (i.e., NCC events that result in the plant being brought to a cold-shutdown condition) have a low frequency of occurrence. The staff is aware of only one such event, which occurred at St. Lucie as discussed above.

This issue was resolved and no new requirements were established because (1) NCC events that result in the plant being brought to a cold-shutdown condition occur infrequently and (2) the actual severity of a specific NCC event will determine the need for actions (if any) and the extent of actions that may be required of any licensee following certain NCC events that may place a reactor vessel in an unanalyzed condition or outside its documented design-basis.

The AP1000 response to this issue in DCD Tier 2 Section 1.9.4.2.3 references DCD Tier 2 Section 3.9.1.1.2.11, and states that the response to GL-92-02 is the responsibility of the COL applicant. The applicant has verified that the analyses to account for NCC events applicable to the AP1000 reactor vessel integrity were evaluated and bounded by the generic assumptions

and conclusions presented in NUREG-1374 and GL 92-02. In DCD Tier 2 Section 3.9.1.1, the applicant presents the AP1000 design transients that are considered in the design and fatigue analysis of ASME Class 1 components. As discussed in Section 3.9.1.1 of this report, all of these transients have been adjusted for a 60-year plant life. In DCD Tier 2 Section 3.9.1.1.2.11, the total number of NCC transients used in the reactor vessel design for its 60-year life span is specified. In addition, in DCD Tier 2 Figure 5.3-3, a generic curve presenting operating temperature, pressure, and cool down rate (not exceeding 100°F/hr) for the reactor vessel is provided, which is consistent with recommendations stated in GL 92-02 and NUREG-1374. On the basis of above information, the staff has concluded that the AP1000 analyses to account for NCC events are bounded by the analyses discussed in NUREG-1374, and are acceptable.

On the basis of the above discussions, the staff concludes that New Generic Issue 79 is resolved for the AP1000 design.

Issue 82: Beyond-Design-Basis Accidents in Spent Fuel Pools

The risks of beyond-design-basis accidents in the spent fuel storage pool were examined in WASH-1400, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975, and it was concluded in the report that these risks were orders of magnitude below those involving the reactor core. Issue 82 in NUREG-0933 reexamined accidents in the spent fuel storage pool for two reasons. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high-density-storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have offered evidence of the possibility of fire propagation between assemblies in an air-cooled environment. These two reasons, in combination, provide the basis for an accident scenario that was not previously considered.

As stated in NUREG-0933, because of the large inherent safety margins in the design and construction of spent fuel pools, this issue was resolved and no new requirements were established.

In DCD Tier 2 Section 1.9.4.2.3, the applicant stated that 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.

The NRC staff reviewed this information provided by the applicant, and the information provided in DCD Tier 2 Section 9.1. As a result of it’s review, the staff concluded Issue 82 is resolved for the AP1000 design.

Issue 83: Control Room Habitability

As discussed in NUREG-0933, Issue 83 addressed the significant discrepancies found during a survey of existing plant control rooms before 1983. These discrepancies included the inconsistencies between the design, construction, and operation of the control room habitability

systems and the descriptions in the licensing-basis documentation. In addition, the staff determined that total system testing was inadequate and that the control systems were not always tested in accordance with the plant TS. Issues related to Issue 83 include (1) Issue B-36, on criteria for air filtration and adsorption units for atmospheric cleanup systems, (2) Issue B-66, on control room infiltration measurements, and (3) Issue III.D.3.4, also on control room habitability. These three issues are discussed in Sections 20.2 and 20.4 of this report.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that habitability of the MCR during normal operation is provided by the non-safety-related nuclear island nonradioactive ventilation system (VBS). In the event of a design-basis accident involving a radiation release or a loss of all ac power event, the non-safety-related nuclear island VBS is automatically terminated, the MCR pressure boundary is isolated, and the the safety-related main control room emergency habitability system (VES) is actuated.

The safety-related VES supplies breathable quality air for the MCR operators while the main control room is isolated. In the event of external smoke or radiation release, the non-safety-related nuclear island VBS provides for a supplemental filtration mode of operation, as discussed in DCD Tier 2 Section 9.4. In the event of a Hi-Hi radiation level, the safety-related VES is actuated. In the unlikely event of a toxic chemical release, the safety-related VES 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 MCR pressure boundary.

In the DSER, the applicant addressed the possibility of toxic gases and substances onsite and offsite affecting control room habitability; the signals, or procedures and operator action for actuation of equipment for control room habitability, and the responsibility of the COL applicant. DCD Tier 2 Section 6.4.7 states that the COL applicant referencing the AP1000 certified design is responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category 1, Class 1E toxic gas monitoring. The applicant committed to comply with RG 1.78-December 2001, Revision 1, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release” to meet the requirements of TMI Action Plan Item IIID.3.4 and GDC 19. In addition, the applicant will identify RG 1.78-December 2001, Revision 1, for DCD Tier 2 Sections 6.4.8, 9.4.13, DCD Tier 2 Table 1.9-1, and DCD Tier 2 Appendix 1A. The NRC Staff expects that Revision 4 of the DCD will reflect DCD Tier 2 sections 6.4.8, 9.4.13, DCD Tier 2 Appendix 1A, and DCD Tier 2 Chapter 16, B3.7.6 accordingly. Therefore, this is Confirmatory Item 6.4-1

The applicant submitted the results of radiological consequence analyses for personnel in the MCR during a DBAs in DCD Tier 2 Section 6.4.4. Details of the analysis assumptions for modeling the doses to MCR personnel were submitted in DCD Tier 2 Section 15.6.5.3. The staff’s review and independent dose assessment will be completed once questions on the assumed aerosol removal rates in the containment, as discussed in unresolved RAIs 470.009 and 470.011, have been resolved. This issue is identified as an Open Item 6.4-1.

The applicant stated in DCD Tier 2 Section 6.4 that the COL applicant will address procedures and training to meet the intent of Issue 83. However, the staff review to conclude that the AP1000 design meets the dose limits of GDC 19 is incomplete, as described above. Therefore, Issue 83 is unresolved for the AP1000 design.

Issue 87: Failure of High-Pressure Coolant Injection Steamline Without Isolation

Issue 87, in NUREG-0933, addressed the staff concerns about a postulated break in the high-pressure coolant injection (HPCI) steam supply line and the uncertainty regarding the operability of the isolation valves for the HPCI steam supply line under these conditions. A break in the line could lead to high flow and high differential pressure that may inhibit closure of the isolation valve. These valves typically cannot be tested in situ for the high design flow rates and pressures. Therefore, subsequent to installation of these valves, it is not feasible to demonstrate the capability of the valves to close when exposed to the forces created by the flow resulting from a postulated break downstream. This issue was resolved by the issuance of GL 89-10 and its supplements on safety-related motor-operated valve (MOV) testing, GL 96-05, and SECY-93-087, which recommended these valves be periodically tested inservice, under full flow and actual plant conditions where practical. Furthermore, in SECY-94-084 and SECY-95-135, additional guidelines are provided for testing MOVs.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that for the AP1000 design, safety-related MOVs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against the maximum differential pressure and flow. The requirements for MOV qualification testing are outlined in DCD Tier 2 Section 5.4.8. In DCD Tier 2 Section 3.9.8.4, the applicant further states that the inservice testing (IST) program for safety-related valves is to be submitted by the COL applicant. This IST program will be developed on the basis of the requirements outlined in DCD Tier 2 Sections 3.9.6 and 5.4.8. The staff concluded that the information related to Issue 87 in the above DCD Tier 2 Sections is acceptable. The staff's evaluation of MOV-related issues is provided in Section 3.9.6 of this report. On the basis of the staff review of this information, Issue 87 is resolved for the AP1000 design.

Issue 89: Stiff Pipe Clamps

NUREG-0933, Issue 89 addressed the staff concerns about the use of structurally stiff clamps for support of safety-related piping systems. Stiff pipe clamp designs differed from conventional pipe support clamps by including features such as uncommonly large dimensions for clamp width and/or thickness, use of high strength or non-ASME approved materials, and large preloading of clamp bolts. The staff's evaluation of this issue found that piping designers commonly assumed that the pipe clamp-induced localized stresses on piping systems were negligible, and did not warrant any specific consideration. This assumption was acceptable for most conventional pipe clamp applications. However, for some applications, certain piping system conditions coupled with the design and installation requirements for stiff pipe clamps could result in interaction effects that should be evaluated in order to determine the significance of any localized stresses induced in the piping. The value/impact assessment included in NUREG-0933 for this issue concluded that it had a low priority ranking for the group of

operating plants considered. However, for future plants, the value/impact assessment resulted in a medium priority ranking for future plants only.

The staff review of DCD Tier 2 Section 1.9.4.2.3 noted that the applicant did not specifically address this issue for the AP1000 design. The staff requested additional information on whether the effects of the use of stiff pipe clamps are considered in the AP1000 piping design. In response to RAI 210.066, the applicant stated that the pipe support design criteria for the AP1000 prohibit the use of “stiff” yoke type pipe clamps, because they induce large local stresses into the supported piping system. The Westinghouse pipe support design criteria document was reviewed by the staff, and based on evaluation of this information, the staff concludes that New Generic Issue 89 is resolved for the AP1000 design.

Issue 93: Steam Binding of Auxiliary Feedwater Pumps

As discussed in NUREG-0933, Issue 93 addressed the potential for a common-mode failure of the auxiliary feedwater system (AFWS) or the emergency feedwater system (EFWS) resulting from steam binding of the AFW pumps caused by heated Main Feed Water (MFW) leaking back through check valves. The AFWS is used to supply water to the SGs should the MFW system be lost, and steam binding of the AFW pumps could result in the loss of the AFWS.

The AFWS may be isolated from the MFW system by a check valve or one or more isolation valves (depending upon the specific design) to keep hot MFW from entering the AFWS. However, operating experience has shown that check valves tend to leak, thus permitting hot MFW to enter the AFWS. This hot feedwater can subsequently flash to steam in the AFW pumps and discharge lines, causing steam binding of the pumps.

In addition, the AFW piping is sometimes arranged so that each AFW pump is connected through a single check valve (which is used to prevent back leakage) to piping that is common to two or three pumps. This arrangement creates the potential for common-mode failures as the hot feedwater leaks back through the check valves into other AFW pumps.

The staff issued GL 88-03 ("Resolution of Generic Safety Issue 93, Steam Binding of Auxiliary Feedwater Pumps," dated February 17, 1988) to the industry as the resolution of this issue. The letter implements monitoring and corrective procedures to minimize the likelihood of steam binding of the AFWS pumps. One of the corrective actions to be taken is the monitoring of AFW pump discharge piping temperatures to ensure that the fluid temperatures remain at or near ambient temperature.

In DCD Tier 2 Section 1.9.4, the applicant states that the AP1000 design does not have a safety-related auxiliary feedwater system. The passive core cooling system is stated to provide the safety-related function of cooling the RCS in the event of loss of feedwater. The startup feedwater system (SUFWS) is stated to provide the SGs with feedwater during startup, hot standby, cooldown, and when the main feedwater pumps are not available, and have no safety-related function other than containment isolation.

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

The staff concluded that steam binding is not a problem for the AP1000 design because the passive core cooling system does not have any pumps that could fail as a result of steam binding, and the SUFWS is not safety-related. Therefore, Issue 93 is resolved for the AP1000 design.

Issue 94: Additional Low-Temperature Overpressure Protection for LWRs

As discussed in NUREG-0933, Issue 94 addressed low-temperature overpressurization events with the resolution of Issue A-26, which is discussed in Section 20.2 of this report. This issue was intended to address the additional guidance for RCS low-temperature overpressure protection (LTOP) to ensure reactor vessel integrity beyond the requirements specified for Issue A-26 in SRP Section 5.2.2, "Overpressure Protection," and BTP RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperature." Issue 94 was resolved with the additional requirements to have the TS for overpressure protection consistent with those specified in Enclosure B to GL 90-06, "Resolution of Generic Issue 70, Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors, Pursuant to 10 CFR 50.54(f)," dated June 25, 1990.

In DCD Subsection 1.9.4.2.3, the applicant states that the reactor vessel for the AP1000 is designed to be less susceptible to brittle fracture during an LTOP event; that material requirements and welding processes are developed to enhance resistance to embrittlement; and that fracture toughness of the reactor vessel is discussed in DCD Subsection 5.3.2.

As discussed in DCD Subsections 1.9.4.2.3 and 5.4.7, one of the safety-related functions of the normal residual heat removal system (RNS) is to provide LTOP for the RCS during refueling, startup, and shutdown operations. The AP1000 RNS design contains a relief valve to provide this safety-related LTOP function. It is designed to limit the RCS pressure within the limits specified in Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50. In accordance with DCD Tier 2 Table 3.2-3 and Figure 5.4-7, this relief valve and its associated piping are classified as safety-related ASME Class 2, seismic Category 1 components. DCD Tier 2 Tables 3.2-1 and 3.9-16 identify these components as being subjected to ISI and testing in accordance with the requirements of the ASME Code, Section XI.

On the basis of the above information, the staff concluded the AP1000 reactor vessel has been adequately designed for LTOP.

GL 90-06 addressed the establishment of additional guidance for RCS LTOP to ensure reactor vessel and RCS integrity beyond that identified in the resolution to Issue A-26, which is discussed in Section 20.2 of this report. As a resolution for Issue 94, GL 90-06 requires a revision to plant TSs for capability of the LTOP system. Other possible solutions identified in

GL 90-06 include hardware modifications including the use of the RHR system relief valves, and requiring the LTOP system to be fully safety-related.

GL 90-06 states that the LTOP availability should be ensured by limiting the allowable outage time to 24 hours for a single LTOP channel while operating in Modes 5 and 6. The AP1000 TS limiting condition for operation (LCO) 3.4.14 for the LTOP system requires that, with the accumulators isolated, either the RNS suction relief valve or the RCS depressurized with an open RCS vent of greater than or equal to 34.8 cm^2 (5.4 in^2) be operable. If the RNS suction relief valve is inoperable, Action Item C of LCO 3.4.14 requires either that the relief valve be restored to operable status or that the RCS be depressurized and the RCS vent be established within 8 hours. The applicant states in BASES B3.4.14 that with the RCS depressurized, a vent size of 34.8 cm^2 (5.4 in^2) is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the maximum pressure on the P/T limit curve. The staff concludes that the AP1000 TS is consistent with GL 90-06, and is acceptable.

Therefore, Issue 94 is resolved for the AP1000 design.

Issue 103: Design for Probable Maximum Precipitation

As discussed in NUREG-0933, Issue 103 addressed the acceptable methodology for determining the design flood level for a particular plant site. The use of the National Oceanic and Atmospheric Administration (NOAA) procedures for determining the probable maximum precipitation for a site was questioned after a licensee disputed the use of two of NOAA's hydrometeorological reports. The issue was resolved with the revisions to SRP Sections 2.4.2 and 2.4.3 in 1989, to incorporate the probable maximum precipitation (PMP) procedures and criteria contained in the latest National Weather Service publications. This was documented in the Federal Register Notice 54 FR 31268 on July 27, 1989, and GL 89-22, "Potential for Increased Roof and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service," dated October 19, 1989.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the PMP is a site-related parameter and the AP1000 is designed for a PMP of 19.4 inches per hour, and 6 inches in a 5 minute interval as specified in DCD Tier 2 Table 2.0-1. The applicant states that the COL applicant has the responsibility to demonstrate that the specific site parameters are within the limits specified for the standard AP1000 design. The specific site is acceptable if the site characteristics are within the AP1000 plant site design parameters in DCD Tier 2 Table 2.0-1. For cases where a specific site characteristic is outside the Table 2.0-1 parameters, the applicant states that the COL applicant must demonstrate that the site characteristic does not exceed the capability of the AP1000 design. Additional information on the site interface parameters is provided in DCD Tier 2 Chapter 2.

The COL applicant must use site-specific environmental data for determining the PMP in accordance with SRP Sections 2.4.2, "Floods," and 2.4.3, "Probable Maximum Flood (PMF) on

Streams and Rivers." This is to ensure the maximum flood level for the AP1000 design specified in DCD Tier 2 Table 2.0-1 shall not be exceeded by the site-specific flood level. This issue is further discussed in Section 2.4 of this report.

Based on review of this information, the staff concludes that New Generic Issue 103 is resolved for the AP1000 design.

Issue 105: Interfacing Systems LOCA at BWRs

Issue 105, in NUREG-0933, was limited to pressure isolation valves (PIVs) in BWRs and was resolved by requiring leak-testing of the check valves that isolate low-pressure systems that are connected at the RCS outside of containment. It is related to Issue 96, which addressed PIVs between the RCS and RHR systems in PWRs. As stated in NUREG-0933, the staff issued Information Notice (IN) 92-36, “Intersystem LOCA Outside Containment," dated May 7, 1992, on this subject. The individual plant examinations required by the staff on operating plants included analyses of these sequences. This issue was resolved without any new requirements for operating plants.

For advanced reactor design, the staff position regarding intersystem LOCA (ISLOCA) protection, as stated in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," as well as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," is that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to full RCS pressure. The phrase "to the extent practicable" is a recognition that all systems must eventually interface with atmosphere, and that it would be difficult or prohibitively expensive to design certain large tanks and heat exchangers with the URS equal to full RCS pressure. Piping runs should be designed to meet the URS criteria, as should all associated flanges, connectors, and packings, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines. The designer should attempt to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

In Section 3.9.3.1 of this report, the staff discusses its evaluation that establishes the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure, should the low-pressure system be subject to full RCS pressure. The subsection within Section 3.9.3.1, "AP1000 Design Criteria for ISLOCA," contains the design criteria proposed by the applicant for the low-pressure portion of the RNS. On the basis of this evaluation, the staff concludes that these criteria are acceptable to assure that the low-pressure side of any applicable system has been designed to meet the full RCS URS criteria.

For all interfacing systems and components that do not meet the full RCS URS criteria, the applicant must justify why it is not practicable to reduce the pressure challenge any further, and also provide compensating isolation capability. For example, applicants should demonstrate for each interface that the degree and quality of isolation or reduced severity of the potential pressure challenges compensate for and justify the safety of the low-pressure interfacing

systems or components. The adequacy of pressure relief and the piping of relief back to primary containment are possible considerations. As identified in SECY-90-016, each of these interfacing systems that has not been designed to withstand full RCS pressure must also include the following protection measures:

(1) the capability for leak testing of the pressure isolation valves

(2) valve position indication that is available in the control room when isolation valve operators are de-energized

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure system and both isolation valves are not closed

DCD Subsection 1.9.5.1.7, “Intersystem LOCA,” provides the applicant’s response regarding compliance of the AP1000 design with the staff position on an ISLOCA. The applicant states that AP1000 has incorporated various design features to address intersystem LOCA challenges. These design features result in very low AP1000 core damage frequency for intersystem LOCAs compared with operating nuclear power plants. The design features are primarily associated with the RNS and are discussed in Section 3 of the WCAP-15993, Revision 1, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated March 2003, and DCD Subsection 5.4.7. WCAP-15993 provides a systematic evaluation of the AP1000 design responses of various systems interfacing the RCS of the ISLOCA challenges.

The systematic evaluation process includes (1) a review of the AP1000 piping and instrumentation diagrams (P&ID) to identify these primary interfacing systems or subsystems directly interfacing with the RCS, and the secondary interfacing systems or subsystems interfacing with the primary interfacing systems, and (2) identification of primary and secondary systems and subsystems having a URS less than the RCS pressure. For those systems or subsystems not meeting the criterion of the URS greater than or equal to the RCS pressure, a design evaluation is made considering whether it is inside containment, whether it meets the three criteria specified in SECY-90-016, and whether it includes other design features specific to them that prevents an ISLOCA to the extent practicable. The report also provides the reasons why it is not practical to design large, low-design pressure tanks and tank structures that are vented to the atmosphere to the high pressure criterion. Interfacing systems or subsystems that connect directly to an atmospheric tank are excluded from further ISLOCA consideration. This is limited to the piping connected directly to the atmospheric tank, up to the first isolation valve other than a locked-open, manual isolation valve.

The staff evaluation of various interfacing systems and subsystems follows.

Normal Residual Heat Removal System

The portions of the RNS from the RCS to the containment isolation valves (CIVs) outside containment are designed to the RCS operating pressure, and the portions downstream of the

CIV and upstream of the discharge line CIV are designed so that its URS is not less than the RCS operating pressure. Therefore, these portions are not of ISLOCA concern. The only portion of the RNS having a URS lower than the RCS pressure is the mechanical shaft seal of the RNS pump, which has a design pressure of 6,200 kPa (900 psig). Subsection 3.1.3.2 of WCAP-15993 discusses the difficulties of designing the RNS pump seal to withstand full RCS pressure. A fundamental problem is that any type of seal that can withstand RCS pressure will likely have abnormally fast wear of the seal faces during normal plant operation at low seal pressures. This increased wear at normal plant operating conditions could well prevent the seal from maintaining the pressure boundary if ever exposed to the full RCS pressure. Use of high pressure seals will also require more frequent maintenance during normal operation. Therefore, it is impractical to design a seal that would maintain the RCS pressure boundary with no leakage, and also operate satisfactorily at low-pressure conditions. The AP1000 RNS pump mechanical seal is designed to minimize the amount of leakage if exposed to full RCS pressure. An Idaho National Engineering Laboratory (INEL) study on the Davis-Besse Nuclear Power Station decay heat removal pump seal, with a design pressure of 3,100 kPa (450 psig), found that the rotating seal would maintain its structural integrity at pressures in excess of 17,200 kPa (2,500 psi), and the mechanical seals can withstand a pressure of 8,300 - 8,600 kPa (1,200 -1250 psi) without leaking. The AP1000 RNS pump mechanical seal is similar to the Davis-Besse decay heat remover (DHR) pumps, but its design pressure is twice as high. The AP1000 RNS pump is fitted with a disaster bushing that limits the leakage from the pump to within the capabilities of the normal makeup system in case of catastrophic mechanical seal failure. Leakage can be controlled with the seal leakoff line routed to a floor drain that is routed to the auxiliary building sump. This is more favorable than a seal specially designed for full RCS pressure at the expense of normal-condition reliability.

In DCD Subsection 5.4.7.2.2, the applicant discusses the AP1000 design features in the RNS specifically aimed at reducing the likelihood of an intersystem LOCA. On the suction side, there is a normally closed, motor-operated isolation valve in the common suction line outside containment, and two normally closed, motor-operated isolation valves in each parallel suction line inside the containment. There is a relief valve with a set pressure of 4,385 kPa (636 psig) connected to the RNS pump suction line inside containment, which is designed to provide low-temperature overpressure protection of the RCS and will reduce the risk of overpressurizing the RNS. On the discharge side, the common discharge line has a safety-related containment isolation check valve inside containment and a safety-related motor-operated isolation valve outside containment. The Motor-operated Valves (MOVs) inside the containment are interlocked to prevent them from opening when the RCS pressure is above the RNS operating pressure of 3,100 kPa (450 psig). The power to these isolation valves is administratively blocked at the valve motor control centers to prevent inadvertent opening. In addition, the discharge header contains a relief valve, which discharges to the WLS effluent holdup tanks, to prevent overpressure in the RNS pump discharge line that could occur if the three check valves and the motor-operated CIV leaked back to the low pressure portions of the RNS.

Also, the RNS design includes an instrumentation channel that indicates pressure in each RNS pump suction line, and a high pressure alarm is provided in the MCR to alert the operator to a condition of rising RCS pressure that could eventually exceed the RNS design pressure. The motor-operated pressure isolation valves also have remote position indications in the MCR. In

addition, these pressure isolation valves are specified in DCD Tier 2 Table 3.9-18 to be subject to TS LCO 3.4.16, which requires the leakage of each RCS PIV to be within limits with leak testing in accordance with surveillance requirement 3.4.16.1. The staff concludes that the RNS design meets the requirements of SECY-90-016.

Chemical and Volume Control System/ Makeup Systems

DCD Subsection 9.3.6 provides a detailed description of the design, functions, and operations of the AP1000 chemical and volume control system (CVS). The purification flow path of the CVS is a high-pressure closed-loop design, which is entirely within the containment and is therefore of no ISLOCA concern. The potential contributors to an ISLOCA are the portions of the CVS located outside the containment, i.e., the letdown line to the liquid radwaste system, and the makeup system.

The CVS makeup pumps operate intermittently to make up for RCS leakage. The pumps start and stop automatically when the pressurizer level reaches the bottom and the top of the normal level band, respectively. The makeup pumps take suction from either the boric acid tank, or the demineralized water storage tank (DWST), and inject into the CVS purification loop return stream. The makeup pumps can also take suction from the waste holdup tanks or the spent fuel pool. The makeup line from the makeup pump discharge to the RCS has a design pressure greater than or equal to the RCS design pressure. However, the pump suction line piping and associated components are low pressure segments, with the URS less than the RCS operating pressure.

In Subsection 3.3.3 of the WCAP-15993, the applicant states that it is not practicable to design the low-pressure portions of the makeup suction piping to higher design pressure. It is not practicable to have a high design pressure for large tanks such as the boric acid tank, which are vented to the atmosphere, as well as the piping directly connected to these atmospheric tanks up to the first isolation valve. The suction lines each contain a check valve that separates the suction piping from a large atmospheric tank. These check valves are designed to open on low differential pressure, and have a high tendency to leak. The suction lines contain relief valves that protect the low-pressure portions of the piping from overpressure in the event of leaking check valves in the discharge line or thermal expansion in case of a loss of miniflow cooling. The relief valves direct any leakage from the discharge line check valves to the WLS effluent holdup tanks (EHTs), which are designed to handle radioactive fluids, and their level is monitored by remote instrumentation.

The passage of the high pressure reactor coolant to the CVS makeup suction is possible only when the makeup pumps are not running, and only if failures or leakage of multiple check valves on the makeup pump discharge side occurs. There is a high-pressure alarm in the pump suction line to alert the operator of overpressurization. In the event of a suction-side overpressurization, the makeup pumps can be operated to terminate overpressurizing the suction piping. If the makeup pumps did not start, the makeup line containment isolation valves would automatically close to terminate the ISLOCA. In addition, the purification loop inlet isolation valves would also be closed on a safeguards actuation signal. These multiple, safety-related isolation valves prevent an ISLOCA in the makeup suction line. As specified in

DCD Tier 2 Table 3.9-16, the purification inlet stop valves, and the purification return line stop valve and check valve are subject to leak testing. These stop valves are provided with position indication in the control room. In addition, the makeup line CIVs also have the capability for leak testing, and are provided with valve position indication in the control room at all times. The staff finds that protection measures meet the intent of SECY-90-016 ISLOCA position.

CVS Letdown/Liquid Radwaste System

The CVS letdown line connects to the high-pressure purification loop inside containment. Immediately downstream of this connection is a high-pressure, multistage letdown orifice, which reduces pressure in the letdown line from the RCS operating pressure to below the design pressure of the low-pressure portion of the letdown line. Around the letdown orifice is a bypass line containing a locked-closed manual isolation valve that is opened only at shutdown when the RCS is depressurized to provide sufficient letdown flow when required. The letdown line is then equipped with two safety-related, normally-closed, fail-closed CIVs where it penetrates containment to the WLS degasifier package and EHTs. The letdown line down to and including the outboard CIV has a design pressure of 17,130 kPa (2485 psig). Downstream of the outboard CIV, the WLS letdown line is a low pressure portion, and therefore does not meet the RCS URS criteria.

In Subsection 3.2.3 of WCAP-15993, the applicant contends that it is not practicable to design the low-pressure portions of the letdown line to a higher design pressure. The WLS EHTs are large atmospheric tanks, and are therefore not practicable for higher design pressure. Nor is the letdown line, which is routed to the degasifier package or the EHTs, and the degasifier package, which discharges directly to the WLS EHTs. The CVS letdown system has the following features to meet the ISLOCA criteria:

- the pressure drop across the CVS letdown orifice protects the WLS from overpressurization during letdown operations by reducing the pressure in the WLS

- in case of an inadvertent valve closure in the WLS during letdown, a relief valve, which discharges directly to the EHT, is provided that would protect the WLS from overpressurization

- due to the letdown orifice, a break in the WLS during letdown from the CVS would result in an RCS leak that is within the capability of the normal makeup system

- if an ISLOCA should occur, it would be terminated by automatic isolation of the two purification loop isolation valves and two letdown isolation valves on low pressurizer level or a safeguards actuation signal

- the letdown line CIVs have the capability for leak testing and have valve position indication in the control room at all times

- the WLS degasifier column contains a high-pressure alarm that would warn the control room operators that the WLS pressure was approaching the design pressure

In addition, as discussed previously, the purification inlet and return stop valves and check valve are subject to leak testing. The staff finds the CVS letdown piping meets the SECY-90-016 ISLOCA position.

Primary Sampling System

The primary sampling system (PSS) collects representative samples of fluids from the RCS and associated auxiliary system process streams, and the containment atmosphere for analysis by the plant operating staff. Section 3.4 of WCAP-15993 provides an ISLOCA evaluation of the PSS. The PSS lines consist of small, 0.64 cm (0.25 in) pipes. The whole PSS is designed to full RCS pressure and temperature, with the exception of the eductor water storage tank (EWST) and its drainage and level indication lines, eductor supply pump seal, and demineralized water supply line. These low-pressure portions have design pressures with URS below the RCS operating pressure. The applicant contends that it is not practical to design the low-pressure portion of the PSS to a higher design pressure because they are at atmospheric pressure and connect to the low-pressure demineralized water transfer and storage system (DWS). Designing the EWST to high pressure to meet ISLOCA criteria would require the DWS to be designed for high pressure, which is not practicable.

The PSS is connected to the RCS through the local sample points in the RCS hot legs, pressurizer vapor and liquid spaces. Each of these sampling connection lines contains a flow-restricting orifice that limits the flow from the RCS in the event of a sample line break, and also reduces the pressure in the sampling lines during sampling operations. Each sampling line also contains a normally closed isolation valve before being connected to a common header. The common header then penetrates the containment with two normally closed CIVs, which are also PIVs and will be isolated on safeguards signal if open for sampling operation. The sampling line then connects to a sample cooler and the sample bottles.

During sampling operations, flow limiting orifices plus ths small diameter of the PSS lines limit flow to approximately 0.5 gpm, and the PSS lines are never pressurized above the design pressure of the low-pressure portions of the PSS. The PSS high pressure-low pressure interface occurs within the grab sample panel, which is a standard panel with design features to prevent backflow and over-pressurization of the low-pressure portions of the system. Even in the unlikely event that over-pressurization would occur, leakage flow from the RCS would be well within the makeup capability of the normally operating makeup system. At any time, the operator would be able to isolate the leakage by closing the PSS CIVs. The CIVs have remote position indication in the control room and are subject to the CIV leakage test. Also, the leakage from these CIVs through the 0.64 cm (0.25 in) pipes would be small. Therefore, the staff finds that the PSS design meets the intent of the ISLOCA criteria.

Solid Radwaste System

The solid radwaste system (WSS), which provides storage facilities for both wet and dry solid wastes prior to and subsequent to processing and packaging, is connected to the high-pressure CVS demineralizers to facilitate transfer of spent resin from the CVS demineralizers to the

spent resin storage tanks (SRSTs). The spent resin header connects to each of the three CVS demineralizers with an individual normally closed isolation valve, and then penetrates containment with two normally closed CIVs to the SRSTs outside. A manual valve is placed downstream of the outboard CIV to isolate the downstream piping to facilitate CIV leak testing. The portion of piping downstream of the manual isolation valve is a low-pressure design with a URS below the RCS operating pressure. Subsection 3.5.2 of WCAP-15993 asserts that it is not practical or necessary to design the WSS to a higher design pressure because the system contains many low-pressure components such as the SRST and resin transfer and mixing pumps.

The WSS spent resin line is normally isolated by locked-closed manual CIVs, which are administratively controlled, have position indications in the control room, and are leak tested in accordance with the inservice testing plan of DCD Tier 2 Subsection 3.9.6. The CVS demineralizers are inside containment and normally circulate reactor coolant at RCS pressure. As such, resin transfer operations are conducted only during refueling operations when the RCS is fully depressurized. During normal power operation, the only pathway to the low-pressure portion of the WSS is for all three closed isolation valves to fail. Should that extremely unlikely event happen, the recirculation loop isolation valves can be closed to isolate the purification loop and the WSS from the RCS. In addition, downstream of the inboard CIV in the resin transfer line, there is a relief valve which discharges to the WLS containment sump inside containment. Therefore, the staff finds that the WSS spent resin lines are not required to be designed to a higher design pressure.

Demineralized Water Transfer and Storage System

The demineralized water transfer and storage system (DWS) receives water from the demineralized water treatment system, and provides a reservoir of demineralized water to supply the condensate storage tank and for distribution throughout the plant. The design and functional details of the DWS are provided in DCD Subsection 9.2.4. The demineralized water transfer pumps take suction from the DWST and supply water through a catalytic oxygen reduction unit to the demineralized water distribution header. From this header, demineralized water is supplied to various systems in the plant. One DWS supply line penetrates containment to a supply header inside containment, which serves as the DWS interface with the PSS and the CVS demineralizers. The DWS provides demineralized water to the PSS to flush the PSS lines prior to RCS sampling, and to the CVS demineralizers to sluice resin to the WSS.

The DWS is a low-pressure system design with a URS below RCS operating pressure. However, the only possible overpressurization pathways from the RCS are the connections to the PSS and the CVS demineralizers inside containment. Overpressurization of the DWS can only occur if there are multiple failures and misalignments of isolation valves and check valves in the high-pressure systems. The DWS supply header inside containment has a relief valve to preclude the possibility of overpressurizing the DWS. In addition, an overpressurization of the DWS would most likely result in the rupture of the DWS header inside containment. Therefore, the staff finds that it is not a concern for ISLOCA.

Summary and Conclusion

The staff concludes that the AP1000 design is consistent with the staff position discussed in SECY-90-016 regarding ISLOCA. Therefore, Issue 105 is resolved for the AP1000 design.

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

NUREG-0933, Issue 106, addressed the release of combustible gases from leaks or pipe breaks resulting in combustible gas accumulation in buildings containing safety-related equipment. NUREG-1364, “Regulatory Analysis for the Resolution of Generic Safety Issue 106: Piping and the Use of Highly Combustible Gases in Vital Areas,” specifically addressed GSI 106, and provided alternatives for prevention, detection, and protection against hazards associated with the release of combustible gases used, stored and piped through safety-related areas and areas that expose safety-related equipment.

As discussed in NUREG-0933 and NUREG-1364, except for hydrogen, most combustible gases are used in limited quantities and for relatively short periods of time. Hydrogen is stored in high-pressure storage vessels and is supplied to various systems in the auxiliary building through small-diameter piping. A leak or break in this piping could result in an explosive mixture of air and hydrogen, posing a potential loss of safety-related equipment.

In DCD Tier 2 Section 1.9.4, Issue 106, the applicant specifies that the AP1000 design uses small amounts of combustible gases for normal plant operations. The plant gas system is discussed in DCD Tier 2 Section 9.3.2. Most such gases are stated to be used in limited quantities and associated with plant functions or activities that do not jeopardize safety-related equipment. These gases are found in areas of the plant that are removed from the nuclear island except the hydrogen supply line to the CVS inside containment, which is the only system on the island that uses hydrogen gas.

Hydrogen gas is supplied to the CVS 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. DCD Tier 2 Section 1.9.4 also specifies that the CVS hydrogen supply piping is routed through the turbine building and into the auxiliary building and then into containment. The hydrogen supply line is routed through the piping/valve room on Elevation 100' of the auxiliary building. The piping valve penetration room in the auxiliary building on Elevation 100' is designed as a 3-hour fire zone. DCD Tier 2 Section 9.3.2 specifies that the hydrogen gas portion of the plant system is a packaged system consisting of a liquid hydrogen storage tank and vaporizer to supply hydrogen gas to the main generator for generator cooling, and to the demineralized water transfer and storage system to support removal of dissolved oxygen, and to other miscellaneous services. The hydrogen supply package system is located outdoors at the hydrogen storage tank area. The turbine building does not house any safety-related systems or equipment. The containment has hydrogen sensors to detect hydrogen leaks. The containment hydrogen concentration monitoring subsystem is designed as Class 1E and seismic Category 1 (DCD Tier 2 Section 6.2.4.1).

The BTP CMEB 9.5-1, Section C.5.d, “Control of Combustibles,” specifies that care should be taken to locate high-pressure storage containers with the long axis parallel to building walls. In addition, BTP 9.5-1 specifies that hydrogen lines in safety-related areas be either designed to seismic Class 1 requirements, or sleeved such that the outer pipe is directly vented to the outside, or should be equipped with excess flow valves so that in case of a line break, the hydrogen concentration in the affected area will not exceed 2 percent. The applicant specified in DCD Tier 2 Table 9.5.1-1, Section C.5.d that the AP1000 design complies with the BTP 9.5.1.

In addition, in DCD Tier 2 Section 9.5.1, the applicant references National Fire Protection Association (NFPA), Standard 50A, “Gaseous Hydrogen Systems at Consumers Sites,” 1999 Edition. DCD Tier 2 Table 9.5.1-3 identifies no exceptions to the referenced NFPA standard. Therefore, on the basis of compliance with the guidance provided in BTP 9.5.1 and the applicable NFPA Standard, GSI 106 is considered resolved.

Issue 113: Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

As discussed in NUREG-0933, Issue 113 addressed the staff's concerns in 1985 that there were no requirements for dynamic qualification testing or surveillance testing of large-bore hydraulic snubbers (LBHSs) (i.e., > 345 MPa (50 kips) load rating). The safety concern was the integrity of the SG lower support structures when subjected to a seismic event. However, this issue was applicable to all SSCs that rely on large-bore hydraulic snubbers for restraint from seismic loads and other dynamic loads, such as water hammer and fluid blowdown due to high-energy line breaks.

LBHSs are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events, yet also allow sufficient piping component flexibility to accommodate system expansion and contraction from such thermal transients as normal plant heatups and cooldowns. Dynamic testing and periodic functional testing are important to verify that the LBHSs are properly designed and maintained for the life of the plant. Issue 113 was resolved with no new requirements, although in a draft RG, SC-708-4, "Qualification and Acceptance Test for Snubbers Used in Systems Important to Safety," the staff provided recommendations for testing hydraulic snubbers used in the design of new plants.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that the AP1000 plant uses significantly fewer hydraulic snubbers than do current operating plants. It further states that, in addition to the recommendations in the NRC draft RG, testing requirements have been established in ANSI/ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants." Because ANSI/ASME OM, Part 4, is referenced as a requirement for IST of snubbers in ASME Section XI, IWF-5000, this is an acceptable commitment for periodic functional testing of LBHSs, and is in accordance with applicable portions of 10 CFR 50.55a(f)(4). DCD Tier 2 Section 3.9.3.4.3 provided a commitment to include dynamic testing as a part of the production operability tests for all snubbers. The production operability tests for LBHSs include: 1) a full service Level D load test to verify sufficient load capacity, 2) testing at full load capacity to verify proper bleed with the control valve closed, 3) testing to verify the control valve closes within the specified velocity range, and 4) testing to demonstrate that breakaway and drag

loads are within the design limits. The staff's evaluation of this issue is discussed further in Section 3.9.3.3 of this report. Based on the staff review of the information provided in DCD Tier 2 relative to periodic functional testing and dynamic qualification testing of LBHSs, the staff concludes that Generic Issue 113 is resolved for the AP1000 design.

Issue 120: On-Line Testability of Protection Systems

As discussed in NUREG-0933, Issue 120 addressed requirements for at-power testing of safety system components without impairing plant operation. The staff raised this issue because it was found in the review of several plant TS in 1985 that some older plants did not provide as complete a degree of on-line testing as other plants then undergoing staff review. The requirements for on-line testing of protection systems are in GDC 21. These requirements apply to both the RPS and the engineered safety features actuation system (ESFAS). A protection system with two-out-of-four (2/4) logic that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three (2/3) logic configuration meets this requirement. This issue was resolved with no new requirements.

Guidance for this issue is provided in RG 1.22, "Periodic Testing of Protection System Actuation Functions," RG 1.118, "Periodic Testing of Electric Power and Protection Systems," and IEEE Standard 338. Conformance to these documents ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is at power without adversely affecting plant operation.

The AP1000 protection system has two-out-of-four (2/4) logic configuration that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three (2/3) logic configuration which meets the requirement in GDC 21 for on-line testing. AP1000 design provision for testing of the protection system is in conformance with RG 1.22 and RG 1.118. The staff concludes that Issue 120 is resolved for the AP1000 design.

Issue 121: Hydrogen Control for Large, Dry PWR Containments

This issue remains open because DCD Tier 2 for the control of combustible gas in containment during accidents does not comply with current regulations.

The NRC has proposed major changes to 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,” and related changes to 10 CFR 50.34 and 10 CFR 52.47, along with the creation of a new rule, 10 CFR 50.46a (see 67 FR 50374, August 2, 2002). These proposed changes are meant to risk-inform the combustible gas control requirements, and constitute significant relaxations of the requirements. The staff plans to finalize the rule changes during 2003.

DCD Tier 2 is written in anticipation of these rule changes. As such, it is not in compliance with the current, more-restrictive regulations. Furthermore, until the proposed rule changes are final and effective, the staff cannot know for certain if the DCD will comply with the revised rule. Therefore, the issue of containment combustible gas control, as well as Issue 121, must remain open at this time. This is an Open Item 6.2.5-1.

Issue 122.2: Initiating Feed and Bleed

As discussed in NUREG-0933, Issue 122.2 investigated the findings of the NRC inspection in 1985 of the loss-of-feedwater event at Davis-Besse on June 9, 1985. The issue dealt with the adequacy of emergency procedures, operator training, and available plant monitoring systems for determining the need to initiate feed-and-bleed cooling following the loss of the SG heat sink (i.e., loss of feedwater). In an analysis of the loss-of-feedwater event, the staff found that operators were hesitant to initiate feed-and-bleed operations, and that the control room instrumentation was inadequate to alert operators to the need to initiate feed and bleed. A loss-of-feedwater in combination with a failure to diagnose and take corrective actions (i.e., initiate feed and bleed) would result in a loss of core cooling.

In DCD Tier 2 Subsection 1.9.4.2.3, the applicant does not address Issue 122.2, except that DCD Tier 2 Table 1.9-2 indicates that the issue was resolved with no new requirements.

However, the Emergency Response Guidelines, in Section 18.9 of this report, is applicable to the AP1000 design. The staff has reviewed the feed-and-bleed emergency guidelines and concluded that they are acceptable. Therefore, Issue 122.2 is resolved for the AP1000 design.

Issue 124: Auxiliary Feedwater System Reliability

Following the loss-of-feedwater event at Davis-Besse plant in 1985, Generic Issue 124 from NUREG-0933 addressed increasing reliability of the auxiliary or emergency feedwater system to 1E-04 unavailability/demand. In 1985, operating experience as well as staff and industry studies indicated that these systems failed at a high rate. The function of the AFWS in the majority of operating plants is to supply feedwater water to the secondary side of the SGs during system fill, normal plant heatup, hot standby, and cold shutdown conditions. The AFWS also functions following loss of normal feedwater flow, including loss resulting from an offsite power failure; additionally, it supplies feedwater to the SGs following accidents such as a main feedwater line break or a main steamline break. Therefore, the reliability of the AFWS is important to plant safety.

The NRC investigation of the Davis-Besse event indicated that the potential inability to remove decay heat from the reactor core was the result of the questionable reliability of the EFWS caused by any or all of the following:

- loss of all EFW as a result of common-mode failure of the pump discharge isolation valves to open

- excessive delay in recovering EFW because of a difficulty in restarting the pump steam-driven turbines once they tripped

- interruption of EFW flow because of failures in steamline break and feedwater line break accident mitigation features

In addition, the investigation of the event indicated that (1) a two-train system with a steam turbine-driven EFW pump may not be able to achieve the desired level of reliability and (2) provisions to automatically isolate EFW from a SG affected by a main steamline or feedwater line break may tend to increase the risk that adequate DHR is not available, rather than decrease it.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that this issue is not applicable to the AP1000 design because the design does not have a safety-related auxiliary feedwater system. The passive core cooling system will provide the safety-related function of cooling the RCS in the event of loss of feedwater to the SGs. The startup feedwater system (SUFWS), which has no safety-related function beyond containment isolation, provides the SGs with feedwater during plant conditions of startup, hot standby, cooldown, and when the main feedwater pumps are not available.

The staff finds that the SUFWS is not a safety-related system and does not have to perform the same safety function as the AFWS and Issue 124 is resolved for the AP1000 design.

Issue 125.II.7: Reevaluate Provisions to Automatically Isolate Feedwater From Steam Generator During a Line Break

As discussed in NUREG-0933, Issue 125.II.7 addressed the long-term actions from NUREG-1154 and the Executive Director for Operations (EDO) memorandum dated August 5, 1985, on the loss-of-feedwater event at Davis-Besse on June 6, 1985. Issue 125.II.7 addressed the need for licensees to reassess the benefits of automatically isolating the EFWS after a break in the secondary side of the SG. For a typical PWR with automatic isolation (AI) of the EFW (AI-EFW), a low-SG-pressure signal causes closure of the main steam isolation valves (MSIVs) and isolation of EFW from the faulted SG during a steamline break. AI-EFW minimizes blowdown from the SG secondary-side line break and limits primary system overcooling and the potential for return to criticality owing to positive moderator reactivity feedwater caused by overcooling of RCS inventory. If the EFW were not isolated, the peak containment pressure for secondary-side breaks would exceed that caused by a large-break LOCA, the design-basis event for the containment.

However, AI-EFW has a disadvantage. If both channels of the controlling isolation logic system were to spontaneously actuate, the availability of EFW would be lost and the MSIVs would close. For the plants using turbine-driven main feedwater pumps, these pumps would be lost following the closure of the MSIVs and the loss of steam, and this loss would result in the loss of the secondary side heat sink. The capability to lock out the isolation logic is necessary to preclude this event.

The staff determined (as stated in NUREG-0933) that, for a new plant, the design does not need to include automatic isolation of EFW following a steamline break or feedwater line break, provided that the results of the analysis of the secondary-side line break and the containment analysis meet the criteria in the appropriate SRP section of NUREG-0800, therefore, Issue 125.II.7 is resolved for the AP1000 design.

Issue 128: Electrical Power Reliability

GSI 128 addresses the reliability of onsite electrical systems and encompasses GSIs 48, 49, and A-30. The staff reviewed the applicant's submittal and concluded that the AP1000 design addresses Issue 48, "LCO for Class 1E Vital Instrument Buses in Operating Reactors"; Issue 49, "Interlocks and LCO for Class 1E Tie breakers"; and Issue A-30, "Adequacy of Safety Related DC Power Supplies" as follows:

- Issue 48 – the applicant provided the LCO in the event of a loss of one or more Class 1E 120 Vac vital instrument buses and associated inverters. The staff finds this LCO acceptable.

- Issue 49 – The AP1000 design does not include Class 1E tie breakers.

- Issue A-30 – The staff evaluated the Class 1E dc distribution system design for the aspects addressed by A-30 in Section 8.3.2.1 of this report and concluded that it is acceptable.

Therefore, Issue 128 is resolved for the AP1000 design.

Issue 130: Essential Service Water Pump Failures at Multi-plant Sites

As discussed in NUREG-0933, Issue 130 addressed the vulnerability of Byron Unit 1 to core-melt sequences in the absence of the availability of Unit 2. While Unit 2 was under construction, it was necessary to make a third service water pump available to Unit 1 via a cross-tie with one of the two Unit 2 essential service water (ESW) pumps. This issue raised concerns relative to multi-plant units that have only two ESW pumps per plant but have cross-tie capabilities. A limited survey of the applicant plants helped to identify the generic applicability of vulnerabilities of multi-plant configurations with only two ESW pumps per plant. In the multi-plant configurations identified (approximately 16 plants), all plants can share ESW pumps via a cross-tie between plants. Additional efforts to resolve this issue included (1) a limited survey of the applicant plants to determine the generic applicability of similar multi-plant configurations with two ESW pumps per plant and whether cross-tie capabilities existed, (2) a survey of B&W, and ABB-CE plants to identify similar multi-plant configurations, and (3) a survey of single-unit plants to determine if similar ESW vulnerabilities existed.

In DCD Tier 2 Section 1.9.4, the applicant states that this issue is not applicable to the AP1000 design because the plant design is for a single independent plant that does not share or cross-tie systems or components with another plant. In DCD Tier 2 Section 3.1.1, the applicant states that if more than one unit is built on the same site, none of the safety-related systems will be shared. The staff finds this acceptable for AP1000 design.

Therefore, Issue 130 is resolved for the AP1000 design.

Issue 135: SG and Steamline Overfill

As discussed in NUREG-0933, Issue 135 was initiated in 1986 to integrate various SG programs and related issues, including water hammer, eddy current testing, and steamline overfill consequences. Overfill is defined as water entering the main steamline caused by excessive feedwater flow resulting from control system failure or a SGTR. This issue was expected to provide a better understanding of SG and secondary-side integrity, including the effects of water hammer on secondary system components and piping, as well as the resultant radiological consequences. Because the staff concluded that SGTR and steamline overfill events are relatively low risks, this issue was resolved and no new requirements were established. This is documented in NUREG-0933 and NUREG/CR-4893, "Technical Findings Report for Generic Issue 135: SG and Steamline Overfill Issues," dated May 1991. A subissue in Issue 135 was the improved eddy current testing of SG tubes. The staff deferred this subissue to the development of a revision to RG 1.83, "Inservice Inspection of Pressurized Water Reactor SG Tubes."

In DCD Tier 2 Section 1.9.4.2.2, the applicant addressed Issue 135 and the four tasks that comprise it, as discussed in NUREG-0933. The evaluation of each task is provided below:

- Task 1 on code and regulatory requirements – DCD Tier 2 Appendix 1A, which discusses the level of conformance with RG 1.83, states that the AP1000 design essentially conforms with the regulatory guidance except where state-of-the-art advances have enhanced inservice inspection techniques. Specifically, as stated in DCD Tier 2 Section 5.4.2.5, the SGs permit access to the tubes for inspection, repair, and plugging per RG 1.83. The AP1000 SGs include features to enhance robotics inspection of tubes without manned entry. As discussed in Sections 5.2.4 and 5.4.2 of this report, the development of the SG tube preservice inspection (PSI) and inservice inspection programs is the responsibility of the COL applicant. SG tube integrity is verified in accordance with this surveillance program as discussed in DCD Tier 2 Section 5.4.15. The programs are plant-specific and will be reviewed by the staff individually for each license application referencing the AP1000 design certification against the staff's regulatory criteria in place at the time of its review. As discussed in the staff’s evaluation of generic safety issue A-3, this action item is designated as COL Action Item 20.2-2.

- Task 2 on SRP Section 15.6.3, "Radiological Consequences of SG Tube Failure" – In DCD Tier 2 Section 15.6.3.1.4, there is a discussion of anticipated operator recovery actions and the effects of those actions in the mitigation of a SGTR event. The automatic SG overfill protection is described in DCD Tier 2 Section 15.6.3.2 and the control logic is described in DCD Tier 2 Section 7.2.

- Task 3 on several generic issues – A compilation of the generic issues are addressed by the following DCD Tier 2 Sections:

    - radiological consequences are discussed in DCD Tier 2 Section 15.6.3
    - SGTR design basis is discussed in DCD Tier 2 Section 15.6.3

  - supplemental tube inspections are discussed in DCD Tier 2 Section 5.4.2.5 and Appendix 1A
  - denting criteria are discussed in DCD Tier 2 Section 5.4.2.4.3
  - safety-related display information is discussed in DCD Tier 2 Section 7.5
  - RCP trip is discussed in DCD Tier 2 Section 7.3.1.1.3.3
  - control room design and design process are discussed in DCD Tier 2 7.5 and 18.8
  - development of EOPs is discussed in DCD Tier 2 18.9
  - organization responses as part of the COL application are discussed in DCD Tier 2 Chapter 13
  - reactor coolant pressure control is discussed in DCD Tier 2 Section 7.7.1.6

- Task 4 on SG overfill, carryover and water hammer – SG overfill, water carryover and water hammer are discussed in DCD Tier 2 15.6.3.2 and the control logic is discussed in DCD Tier 2 7.2 .

Therefore, Issue 135 is resolved for the AP1000 design.

Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits

As discussed in NUREG-0933, Issue 142 addressed observations in 1987 during safety parameter display system (SPDS) evaluation tests that, for electrical transients below maximum credible levels, a relatively high level of noise could pass through types of isolation devices and be transmitted to safety-related circuitry. In some cases, the amount of energy transmitted through the isolator could damage or seriously degrade the performance of the Class 1E components; in other cases, the electrically generated noise on the circuit may cause the isolation device to give a false output. This issue addressed electrical isolators used to maintain electrical separation between Class 1E and non-Class 1E electrical systems and prevent malfunctions in the non-Class 1E circuits from degrading the performance of Class 1E circuits.

In resolving this issue, the staff determined from operating experience that isolation devices perform satisfactorily in the operating environment and have not been exposed to failure mechanisms that resulted in signal leakage. This determination was made, however, on the basis that current plants predominantly use electromechanical controls and may not be applicable to instrumentation and control (I&C) systems with digital or electronic components. This issue was resolved with no new requirements established.

The use of isolation devices in the AP1000 instrumentation and control architecture is described in DCD Tier 2 Sections 7.1.2.11, “Isolation Devices,” 7.1.4.2.7, “Conformance to the requirements Concerning Control and Protection System Interaction,” and 7.7.1.11, “Diverse Actuation System.” The isolation devices are tested to conform to design requirements and this testing will identify the devices potentially susceptible to electrical leakage. The applicant further stated that 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 COL license holder. This is COL Action Item 20.3-1.

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 the DCD Tier 2 Section 7.1.2.9, “Intercabinet Communications.” The electromagnetic design, testing, and qualification is performed as described in DCD Tier 2 Section 7.1.4.1.6, “Design Basis: Protection Against Natural Phenomena and Unusual Events.”

The diverse actuation system (DAS), which is described in DCD Tier 2 Section 7.7.1.11, uses sensors that are separate from those being used by the protection system and the control system. This prohibits failures from propagating to the other plant systems through the use of shared sensors.

Based on the above discussion, with the COL Action Item 20.3-1, the staff considers that Issue 142 is resolved for the AP1000 design.

Issue 143: Availability of Chilled-Water Systems and Room Cooling

As discussed in NUREG-0933, Issue 143 addresses problems experienced in recent years at several nuclear plants with safety system components and control systems that have resulted from a partial or total loss of HVAC systems. Many of these problems exist because of (1) the desire to provide increased fire protection and (2) the need to avoid severe temperature changes in equipment control circuits. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection of equipment required for safe shutdown. Generally, this improvement has been accomplished by enclosing the affected equipment in small, isolated rooms. However, the result has been a significant increase in the impact of the loss of room cooling. Another problem resulting from loss of room cooling is the advancement in control circuit design. With the introduction of electronic integrated circuits, plant control and safety have improved; however, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as RHR pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once these failures occur, the Advisory Committee on Reactor Safeguards believed that the impact of these failures on the proper functioning of air cooling systems has not been reflected in the final PRAs of plants. Thus, plants with similar, inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled-water systems to remove heat from the rooms containing the components. If chilled-water and HVAC systems are unavailable to remove heat, the ability of the safety equipment within the rooms to operate as intended cannot be assured.

Issue 143 has not been generically resolved and is classified in NUREG-0933 as a high-safety priority. A possible solution to this issue would require a reevaluation of each plant's room heat

load and heatup rate in order to locate areas in which the dependence of equipment operability on HVAC and room cooling may be reduced. Although the total elimination of this dependence may not be possible at all plants, this analysis would locate areas in which this dependence is critical. The critical dependencies and the ability to reduce them could be determined through the use of a plant-specific PRA. After the critical dependencies are identified, each plant would implement procedural changes (to provide alternate cooling) to eliminate or reduce the dependencies where possible. Hardware modifications may be needed for situations in which a procedure change cannot be implemented to reduce a critical dependency.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that this issue does not apply to the AP1000 design because the design 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. The staff agrees with this statement. Therefore, Issue 143 is resolved for the AP1000 design.

Issue 153: Loss of Essential Service Water (ESW) in LWRs

As discussed in NUREG-0933, Issue 153 addressed the reliability of ESW systems and related operating problems. In a comprehensive NRC evaluation of operating experience related to ESW systems (NUREG-1275, Volume 3, "Operating Experience Feedback Report," dated November 1988), a total of 980 operational events involving the ESW system were identified, of which 12 resulted in complete loss of the ESW system. Among the causes of failure and degradation are (1) various fouling mechanisms (sediment deposition, biofouling, corrosion and erosion, foreign material and debris intrusion); (2) ice effects; (3) single-failures and other design deficiencies; (4) flooding; (5) multiple equipment failures; and (6) human and procedural errors.

At each plant, the ESW system supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink. The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate the consequences of the accident. Under normal operating conditions, the ESW system provides component and room cooling. During shutdowns, it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to the fire protection system, cooling towers, and water-treatment systems at a plant.

The design of the ESW system varies substantially from plant to plant and the ESW system is highly dependent on the nuclear steam supply system (NSSS). As a result, generic solutions (if needed) are likely to be different for PWRs and BWRs. The possible solutions are (1) installation of a redundant intake structure including a service water pump, (2) hardware changes to the ESW system, (3) installation of a dedicated RCP seal cooling system, or (4) changes to TS or operational procedures.

In the resolution of Issue 130 on ESW pump failures at multi-plant sites, discussed earlier in this section, the staff surveyed seven multi-plant sites and found that loss of the ESW system could be a significant contributor to core-damage frequency. The generic safety insights

gained from this study supported previous perceptions that ESW system configurations at other multi-plant and single-plant sites may also be significant contributors to plant risk and should also be evaluated. As a result, Issue 153 was identified to address all potential causes of ESW system unavailability, except those that had been resolved by implementation of the requirements in GL 89-13.

The staff resolved Issue 153 with no new requirements established for operating and new plants.

In DCD Tier 2 Section 1.9.4.2.3, the applicant states that this issue does not apply to the AP1000 design because the design does not rely on the service water and component cooling water systems to provide safety-related safe shutdown. The staff agrees with this statement. Therefore, Issue 153 is resolved for the AP1000 design.

Issue 163: Multiple SG Tube Leakage

Issue 163, “Multiple SG Tube Leakage,” in NUREG-0933 identified a safety concern associated with potential multiple SG tube leaks triggered by a main steamline break (MSLB) outside containment that cannot be isolated. This sequence of event could lead to core damage that could result from the loss of all primary system coolant and safety injection fluid in the refueling water storage tank. The NRC has given Issue 163 a HIGH priority ranking, and is working toward a resolution of the issue.

DCD Tier 2 Section 1.9, Compliance with Regulatory Criteria, does not address this issue, except that Table 1.9-2 indicates that Generic Issue 163 is unresolved pending generic resolution. In response to a staff question (RAI 440.184, Revision 1, the applicant letter DCP/NRC1566, April 7, 2003), the applicant stated the issue should be considered closed for the AP1000 based on the following evaluation.

The AP1000 plant response to a MSLB scrams the reactor automatically and removes decay heat via the intact SG or the passive residual heat removal heat exchanger (PRHRHX). If the MSLB is not isolated, the RCS will continue to lose coolant after shutdown through leaking SG tubes, and the plant responds to the scenario as for a small-break LOCA. The core makeup tanks (CMTs) 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 to the containment pressure, the much lower containment pressure stops the leakage through through the leaking SG tubes. Therefore, no long-term core uncovery is expected. Also, the elevation of the high point of the steam line is approximately 80 feet higher than the elevation of the ADS-4 discharge. Therefore, once the ADS-4 is actuated and the RCS depressurized, the leakage from the primary side through the SG tubes will stop. Based on this analysis, the applicant concludes that the ADS-4 operation would reduce any postulated primary-to-secondary leakage for a hypothetical MSLB followed by SG tube leakage.

The staff agrees that the issue should be closed for the AP1000 design. The concern of Generic Issue 163 was that the multiple SGTR as a result of the MSLB and degarded SG tubes

could result in core damage due to depletion of the reactor coolant and safety injection fluid in the refueling water storage tank. For the AP1000 design, a SGTR is mitigated using the passive core cooling system (PXS), initially through the PRHRHX, and CMTs. After the CMTs drain to the low level to actuate the ADS, the RCS depressurization would result in the gravity injection from the IRWST, and eventually from the containment recirculation. The scenario that the safety injection from the RWST, which is outside the containment in the existing plants, will be depleted to result in core damage is not likely for the AP1000 design as the IRWST and containment recirculation will continue to provide core cooling.

Since the resolution of GI 163 is an ongoing NRC effort, any future requirements for the resolutions of this issue will be required of the COL applicant if applicable to the AP1000 design.

Issue 168: “Equipment Qualification of Electric Equipment”

This issue is related to the effects of cable aging and whether the licensing basis for older plants should be reassessed or enhanced in connections with license renewal, or whether they should be reassessed for the current license term. This issue is not applicable to the AP1000 design and combined license actions on the AP1000 will be based upon current cable requirements. Therefore, reassessments are not required for AP1000.

Issue 185: Control of Recriticality Following SBLOCA in PWRs

As discussed in NUREG-0933, issue 185 addressed the possibility of a re-criticality because of potential unborated water slug entering the core following a small break LOCA event. Specifically, the issue was identified following an Nuclear Reactor Regulation (NRR) request for reconsideration of the safety priority ranking of GSI-22, “Inadvertent Boron Dilution Events,” based on new information on high burn-up fuel and new calculations provided by the B&W Owner’s Group (B&WOG). In particular, reactivity insertion tests conducted on high-burnup fuel have indicated that high burn-up fuel may be more susceptible to reactivity events than previously expected. In addition, calculations conducted by the B&WOG have predicted that prompt critically is possible, and that significant heat generation under these conditions may result from small-break LOCAs.

The applicant has addressed this issue in consideration of the AP1000 design as described in its response to RAI 440.099, Revision 1.

As described in Section 15.2.8 of this report, the staff completed its review of the small-break LOCA deboration issue for the AP1000 design and concluded that the AP1000 design is acceptable with respect to the deboration issue. Therefore, the staff considers issue 185 resolved for the AP1000 design.

Issue 189: Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident

This issue is primarily concerned with the fact that the hydrogen igniters in ice condenser and BWR Mark III containment plants are not supplied with emergency power and would not function during SBOs. There is a potential, then, that such a containment might fail due to uncontrolled hydrogen combustion during an accident with a SBO. At this time, all of the operating plants that have hydrogen igniters (and so are susceptible to this weakness) have either ice condenser or Mark III containments. Although the AP1000 does not have such a containment, it does have hydrogen igniters, and so it is prudent for the staff to consider the applicability of GSI-189 to AP1000.

In DCD Tier 2 Table 1.9-2, the applicant identified GSI-189 as either having a priority of "low, drop" or as not having been prioritized. The staff agrees with this assessment because:

- As a result of the passive design, the fraction of core damage frequency that involves station blackout is less than 0.01 (1 percent). Thus, the igniters are highly reliable.

- Despite the low contribution to core damage figure (CDF) from station blackout, AP1000 has the capability to power the igniters from non-safety-grade diesel generators or station batteries in the event of SBO.

Therefore, the staff concludes that GSI-189 is resolved for the AP1000 design.

Issue 191: Assessment of Debris Accumulation on PWR Sump Performance

Similarly to Issue A-43 (see above description), Issue 191 concerns the potential for debris blockage to interfere with the capability of the recirculation mode of the ECCS to provide long-term reactor core cooling at PWRs. Although Issue A-43 was considered resolved in 1985, later operational events at BWRs and confirmatory testing demonstrated that its resolution was not based on a complete understanding of debris generation, transport, and head loss. Thus, during the resolution of the clogging issue for BWRs, Issue 191 was opened to re-examine the effect of debris blockage on PWR sump performance in a more accurate manner.

At the present time, the NRC is in the process of resolving Issue 191 for the current generation of PWRs, and some part of research and analysis is incomplete. Section 6.2.1.8 of this report provides the staff’s further evaluation of the AP1000 suction screens in accordance with the current state of knowledge regarding Issue 191. This issue is still open for AP1000.

20.4 Three Mile Island Action Items

Issue I.A.1.4: Long-Term Upgrading of Operating Personnel and Staffing

As discussed in NUREG-0933, Issue I.A.1.4 addressed changes to 10 CFR 50.54, "Conditions of Licenses," concerning shift staffing and working hours of licensed operators. The final rule that amended 10 CFR 50.54 was approved on April 28, 1983. This issue was resolved and

new requirements were established. The applicant did not address this issue in its DCD Tier 2. It concluded, in Table 1.9-2, that this issue was not relevant to the AP1000 design because this issue is not a design certification issue, but is the responsibility of the COL applicant.

The staff, however, considers this issue not relevant to the AP1000 design because it is an operational issue outside the scope of AP1000 design certification. The organizational structure of the site operator is discussed in Section 13.1 of this report. The COL applicant will be responsible for addressing this issue as part of the licensing process and is COL Action Item 20.4-1.

Issue I.A.2.6(1): Revise RG 1.8

As discussed in NUREG-0933, Item I.A.2.6(1) addressed the revision of RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," following the publication of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The revisions to the RG were to address acceptable means to meet new requirements for long-term upgrading of training and qualifications for operational personnel. The revisions to RG 1.8 were approved by the Commission and published in May 1987 (Federal Register Notice 52 FR 16007). This issue is resolved with new requirements established.

The staff considers this issue not relevant to the AP1000 design because it is an operational issue outside the scope of the design certification. The organizational structure of the site operator is discussed in Section 13.1 of this report. The COL applicant will be responsible for addressing this issue as part of the licensing process and is COL Action Item 20.4-2.

Issue I.A.4.1(2): Interim Changes in Training Simulators

As discussed in NUREG-0933, Issue I.A.4.1(2) addressed the specific training simulator weaknesses identified in the short-term study, of Issue I.A.4.1(1), NUREG/CR-1482, "Nuclear Power Plant Simulators: Their Use in Operator Training and Requalification," dated August 1980. This issue was resolved with the revision to RG 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," in April 1981, and new acceptance requirements were established.

The applicant did not address this issue in the DCD Tier 2. It concluded, in Table 1.9-2, that this issue was not relevant to the AP1000 design because this issue is not a design certification issue, but is the responsibility of the COL applicant. The staff also considers this issue not relevant to the AP1000 design because it is an operational issue outside the scope of the design certification. Training materials are discussed in Section 13.2 of this report. The COL applicant will be responsible for addressing this issue as part of the COL process and is part of COL Action Item 20.4-2. Therefore, Issue 1.A.4.a(2) is resolved for the AP1000 design.

Issue I.A.4.2: Long-Term Training Simulator Upgrade

As discussed in NUREG-0933, Issue I.A.4.2 addressed the capabilities of training simulators. This issue was resolved by Revision 1 to RG 1.149 ("Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations"), 10 CFR 55.45(b) on approved or certified simulation facility in licensed operator operating tests, and NUREG-1258 ("Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55," dated December 1987). New requirements were established. The applicant did not address this issue in the DCD Tier 2. It concluded, in Table 1.9-2, that this issue was not relevant to the AP1000 design because this issue is not a design certification issue, but is the responsibility of the COL applicant. This is part of COL Action Item 20.4-2. As indicated in DCD Tier 2 18.3, Element 2, Operating Experience Review, of this report, the applicant, in WCAP-14645, has satisfactorily addressed this item. Therefore, Issue I.A.4.2 is resolved for the AP1000 design.

Issue I.C.1: Guidance for Evaluation and Development of Procedures for Transients and Accidents

As discussed in NUREG-0933, Issue I.C.1 addressed the preparation of emergency operating procedures (EOPs). The information in EOPs should provide assurance that operator and staff actions are technically correct and the procedures are easily understood for normal, transient, and accident conditions. The EOPs must be function-oriented procedures to mitigate the consequences of the broad range of events and subsequent multiple failure or operator errors, without the need to diagnose specific events. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing and surveillance must be in compliance with the guidance provided in NUREG-0737 and its Supplement 1.

The applicant did not address this issue in DCD Tier 2 information. It concluded, in Table 1.9-2, that this issue is not an AP1000 design certification issue, but is the responsibility of the COL applicant. The staff has identified COL Action Item 18.9-2 that the COL applicant should develop plant-specific EOPs using the guidance provided by the emergency response guidances (ERGs).

DCD Tier 2 Section 18.9.8 summarizes high level operator actions associated with transients and accidents. In this Section, the applicant provides the descriptions of methodologies used to develop the ERGs and an outline of high-level mitigation strategies.

There are fundamental differences between the Low Pressure (LP) reference plant and AP1000 design in the safety system design, operation, and philosophy of emergency mitigation and recovery. Unlike the LP reference plant where the safety systems are active systems, the safety systems in AP1000 are passive systems. Active systems are non-safety-related systems providing defense-in-depth functions. Even though the passive safety systems perform similar functions as the active safety systems in the LP reference plants, the AP1000 mitigation sequences, including the actuation of the active defense-in-depth systems and passive safety systems, and plant conditions at which these systems will be actuated and will remain operating, differ from the LP reference plants. For AP1000, the active systems, though not

actuated by safeguard signals, are manually actuated and relied upon as first line of defense to avoid unnecessary actuation of passive safety systems.

The plant responses, including possible adverse systems interactions between the active and passive systems, may also differ significantly from the LP reference plants. Certain issues where operator actions play key roles in the accident scenarios require the AP1000-specific ERGs as a basis for resolution. For example, in a SGTR event, operator's actions to isolate the faulted SG and other mitigation and recovery actions to minimize the possibility of radioactive releases through the main steam safety valves will be important for the resolution of the issue of containment bypass resulting from a SGTR event. Additionally, the ERGs should include guidance for low power and shutdown operations, when many systems will be out for maintenance and the plant is in a configuration different from the normal operation, and for severe accident management.

To satisfy these requirements, the staff considered the need for the ERGs and supporting analyses necessary to demonstrate the effectiveness of operator actions in response to transients and accidents. As indicated in DCD Tier 2 18.3, Element 2, Operating Experience Review, of this report, the applicant, in WCAP-14645, Revision 2, "Human Factors Engineering Operating Experience Review Report for the AP600 Nuclear Power Plant," dated December 1996, has satisfactorily demonstrated the effectiveness of operator actions in response to transients and accidents. Therefore, Issue I.C1 is resolved for the AP1000 design, based on the COL Action Item 18.9-2.

Issue I.C.5: Procedures for Feedback of Operating Experience to Plant Staff

As discussed in NUREG-0933, Issue I.C.5 addressed the quality of procedures for feedback of experience at operating plants. This issue was clarified in NUREG-0737 and requirements were issued there.

In DCD Tier 2 Section 1.9.3, Item (3)(i), the applicant addresses this issue and states that 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 licensees, Federal Register information, and NRC GLs. It is stated that lessons-learned experience has been incorporated into the AP1000 design through the participation in developing Volume III of the ALWR Utility Requirements Document (URD) and in ALWR Utility Steering Committee activities.

The applicant addressed the responsibility of the designer of the plant; however, the COL applicant will also be responsible for site-specific information at the COL and operational phases. Development of detailed procedures is outside the scope of the AP1000 design certification and is the responsibility of the COL applicant. This is part of thr COL Action Item 20.4-2. Therefore, Issue I.C.5 is resolved for the AP1000 design.

Issue I.C.9: Long-Term Program for Upgrading Procedures

As discussed in NUREG-0933, Issue I.C.9 addressed the upgrading of procedures at operating plants. With the exception of EOPs, this issue was clarified in Supplement 1 of NUREG-0737 and resolved with Revision 1 of SRP Section 13.5.2. This issue was resolved with no new requirements.

The applicant did not address this issue in the DCD Tier 2. It concluded, in Table 1.9-2, that this issue was not relevant to the AP1000 design because this issue was resolved with no new requirements.

However, the responsibility of the COL applicant in procedure development should be identified. The methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures should be addressed.

This is a COL Action Item in DCD Tier 2, 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. Issue I.C.9 is resolved for the AP1000 design.

Issue I.D.1: Control Room Design Reviews

As discussed in NUREG-0933, Issue I.D.1 addressed licensees performing a detailed review of their control room using human factors engineering (HFE) techniques and guidelines to identify and correct design deficiencies. This issue was clarified in NUREG-0737 and NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981, and requirements were issued. This issue is considered resolved.

In DCD Tier 2 1.9.3, Item (2)(iii), the applicant states that the AP1000 MCR was designed by a multi-disciplined man-machine interface design team using state-of-the-art human factors principles. The team used a control room design process predicated on the functional decomposition of the plant, integrating the capabilities of both man and machine. DCD Tier 2 Chapter 18 discusses the MCR design process and DCD Tier 2 1.9.1 provides information on the conformance of the design with applicable RGs.

As indicated in DCD Tier 2 18.3, Element 2, Operating Experience Review, the applicant satisfactorily addressed this item in WCAP-14645. Therefore, Issue I.D.1 is resolved for the AP1000 design.

Issue I.D.2: Plant Safety Parameter Display Console

As discussed in NUREG-0933, Issue I.D.2 addressed improving the presentation of the information provided to control room operators. The requirements for this issue are in Supplement 1 to NUREG-0737. This issue raised the need for a SPDS that clearly displays a minimum set of parameters defining the safety status of the plant. Paragraph (2)(iv) of 10 CFR 50.34(f) requires a plant SPDS console that will provide such a display to operators, and that is

capable of displaying a full range of important plant parameters and data trends on demand and indicating when process limits are being approached or exceeded.

In DCD Tier 2 Section 1.9.3, Item (2)(iv), the applicant states that the purpose of the plant SPDS is to display the important plant variables in the MCR to assist the operator in rapidly and reliably determining the safety status of the plant.

The SPDS design is discussed in DCD Tier 2 Chapter 18. The SPDS requirements are stated to be specified during the MCR design process, discussed in Issue I.D.1, and are met by the MCR design, specifically as part of the alarms, displays, and controls. The requirements 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 SPDS requirement for presentation of plant data in an analog fashion before reactor trip is met by the design of the graphic cathode ray tube (CRT) displays. Displays are available at the operator workstations, the supervisor workstation, the remote shutdown workstation, and at the technical support center.

As indicated in DCD Tier 2 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. Issue I.D.2 is resolved for the AP1000 design.

Issue I.D.3: Safety System Status Indication

As discussed in NUREG-0933, Issue I.D.3 addressed the need for those licensees and applicants who have not committed to RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems,” to install a bypass and inoperable status indication system to give operators timely information on the status of the safety systems. Resolution of this issue requires adoption of the guidelines in RG 1.47.

In DCD Tier 2 1.9.3, Item (2)(v), the applicant states that the AP1000 main control room meets the NRC RG 1.47 recommendations, including automatic indication of bypassed and inoperable status of plant safety systems. This is described in DCD Tier 2 Chapters 7 and 18, and 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.

The AP1000 design incorporates this information into the alarm system, the operator's workstation, and wall panel information system in the main control room. High level plant status during any plant state is continuously available on the wall panel information system. At the

operator's workstation, physical and functional displays show how a component's availability or unavailability impacts the alignment and availability of the system. This is indicated on the display that includes the bypassed or deliberately induced inoperability of the protection system and the systems actuated or controlled by that protection system. Alarms on the operator's workstation and the wall panel information system indicate abnormal conditions. Improper safety system alignments, safety-related component unavailability, and bypassed protective functions are considered in the alarm logic. This information is continuously monitored by the alarm system.

On the basis of the above information, the staff concludes that the AP1000 design meets the guidelines in RG 1.47 and, therefore, meets the requirements of Issue I.D.3 with respect to the I&C design for safety system status monitoring and is resolved for AP1000.

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

As discussed in NUREG-0933, Issue I.D.5(2) addressed the need to improve the operators' ability to prevent, diagnose, and properly respond to accidents. This issue was originally raised in 1980, in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," dated May 1980, and led to new NRC requirements. Guidance for addressing the issue is in RG 1.47, which describes an acceptable method for implementing the requirements of IEEE 279-1971 ("Criteria for Protection Systems for Nuclear Power Generating Stations") and Appendix B (Criterion XIV) of 10 CFR Part 50, with respect to the bypass or inoperable status of safety systems; and RG 1.97, which defines an acceptable method for implementing NRC requirements to provide instrumentation and to monitor plant variables and systems during and following an accident.

The acceptance criteria for the resolution of this issue are:

- For Engineered Safety Feature (ESF) status monitoring, RG 1.47 recommends automatic bypassed or inoperable status indication at the system level for plant protection systems, safety systems actuated or controlled by protection systems, and their auxiliary and supporting systems. These features should indicate in the control room and should have manual input capability.

- For post-accident monitoring instrumentation, RG 1.97, Revision 2, gives criteria for design and qualification of the instrumentation. Three categories (designated 1, 2, and 3) provide a graded approach to requirements on the basis of the importance to safety of the variable being monitored. Criteria exist for equipment qualification, redundancy, power sources, channel availability, quality assurance (QA), display and recording range, equipment identification, interfaces, servicing, testing and calibration, human factors, and direct measurement. The actual variables to be monitored are tabulated by type, and the instrumentation design and qualification requirement (Category 1, 2, or 3) are identified for each variable.

In DCD Tier 2 Section 1.9.4.2.1, item I.D.5(2), the applicant states that the AP1000 design conforms to and meets the intent of RG 1.97, which provides acceptable guidance 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. For the AP1000 design, an analysis was conducted to identify the appropriate plant variables, and establish the appropriate design-basis and qualification criteria for instrumentation used by an operator for monitoring conditions in the RCS, secondary heat removal system, the containment, and the systems used for attaining a safe-shutdown condition. This is discussed in DCD Tier 2 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 to satisfy RG 1.97 are processed and displayed by the qualified data processing system (QDPS) discussed in DCD Tier 2 Section 18.8 . The verification and validation (V&V) of the QDPS complies with the V&V process described in DCD Tier 2 Section 18.11.

In DCD Tier 2 Section 7.5, the applicant compares the AP1000 design against the criteria in Revision 3 of RG 1.97 and addresses accident monitoring instrumentation. The design complies with Revision 3 of RG 1.97.

Issue I.D.5(2) was resolved with the issuance of Revision 2 of RG 1.97. On the basis of the information provided by the applicant and the fact that the AP1000 design is in compliance with Revision 3 of RG 1.97, the staff concludes that this issue has been addressed. Therefore, Issue I.D.5(2) is resolved for the AP1000 design.

Issue I.D.5(3): On-Line Reactor Surveillance Systems

As discussed in NUREG-0933, Issue I.D.5(3) addressed the benefit to plant safety and operations of continuous on-line automated surveillance systems. Systems that automatically monitor reactor performance can benefit plant operations and safety by providing continuous diagnostic information to the control room operators, to predict anomalous plant behavior.

Various methods of on-line reactor surveillance have been used, including neutron noise-monitoring in BWRs to detect vibrations in internal components, and pressure noise surveillance at TMI-2 to monitor primary loop degasification. On-line surveillance data have been used to assess loose thermal shields.

In DCD Tier 2 Section 1.9.4.2.1, the applicant states that the AP1000 Reactor Coolant Pressure Boundary (RCPB) is monitored for leaks from the reactor coolant and associated system by a variety of components located in multiple systems, and that the leak detection system is designed according to the requirements of 10 CFR Part 50, Appendix A, GDC 30. The applicant also states that a Digital Metal Impact Monitoring System (DMIMS) monitors the Reactor Coolant System (RCS) for the presence of loose metallic parts, and that this system conforms with the guidance provided in RG 1.133, Rev. 1.

The acceptance criteria for leak monitoring are in RG 1.45, which documents acceptable methods for channel separation, leakage detection, detection sensitivity and response time,

signal calibration, and seismic qualification of RCPB leakage detection systems. It defines the regulatory position for an acceptable design of these systems.

The acceptance criteria for loose-parts monitoring are in RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." This RG gives guidelines on such system characteristics as sensitivity, channel separation, data acquisition, and seismic and environmental conditions for operability. It also identifies alert levels, data acquisition modes, safety analysis reports, and TS pertaining to a loose-parts monitoring system.

DCD Tier 2 Section 5.2.5 provides detailed discussion of the detection of leakage through RCPB. Section 5.2.5 of this report describes the staff evaluation of the AP1000 RCBP leakage detection. DCD Tier 2 Section 4.4.6.4 describes the AP1000 DMIMS. Section 4.4.4.2 of this report discusses the staff's evaluation of the AP1000 DMIMS.

Based on its evaluations discussed in Sections 5.2.5 and 4.4.6.4, the staff concludes that Issue I.D.5(3) is resolved for the AP1000 design.

Issue I.F.1: Expanded Quality Assurance List

10 CFR 52.47(a)(ii) requires that an applicant for design certification include a demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f). 10 CFR 50.34(f)(3)(ii) requires that an application shall provide sufficient information to demonstrate that the following requirement has been met: "ensure that the quality assurance (QA) list required by Criterion II, App B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)."

This requirement was intended to expand the QA list to ensure that non-safety related SSCs that were important to safety were subject to appropriate quality assurance controls. In reviewing the AP1000 design certification application, the staff determined that other quality programs, such as the reliability assurance program and the regulatory treatment of non-safety-related systems (RTNSS), are sufficient to provide reasonable assurance that non-safety-related SSCs that are important to safety will perform satisfactorily in service.

The staff evaluation of the AP1000 reliability assurance program is discussed in Section 17.4 and the staff evaluation of the RTNSS program is discussed in Section 22 of this report.

Based on the existence of alternate quality programs that provide reasonable assurance that non-safety-related SSCs important to safety will perform satisfactorily in service, the staff concluded that the requirements of 10 CFR 50.34(f)(3)(ii) are not technically relevant to the AP1000 design certification. Therefore, the requirement to include all SSCs important to safety in the QA list is not applicable to the AP1000 design certification.

Issue I.F.2: Develop More Detailed Quality Assurance Criteria

10 CFR 52.47(a)(ii) requires that an applicant for design certification include a demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth

in 10 CFR 50.34(f). 10 CFR 50.34(f)(3)(iii) states, in part, that an application shall provide sufficient information to demonstrate that the following requirements have been met:

> Establish a quality assurance (QA) 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 QA personnel in the documented review of and concurrence in quality related procedures associated with the design, construction and installation; (D) establishing criteria for determining QA Programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as built" documentation; and (H) providing a QA role in design and analysis activities.

The requirements in 10 CFR 50.34(f)(3)(iii) were intended to improve the QA program to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety. The NRC staff reviewed the requirements of 10 CFR 50.34(f)(3)(iii) to determine which requirements were technically relevant to a design certification applicant. The NRC staff determined that the requirements contained in 10 CFR 50.34(f)(3)(iii)(B) were associated with QA activities during plant construction and therefore were not technically relevant to a design certification applicant. Similarly, the requirements of 10 CFR 50.34(f)(3)(iii)(G) were associated with control of "as-built" documentation and were therefore not technically relevant to design certification. In DCD Tier 2 Section 1.9.3, the applicant indicated that the AP1000 quality assurance plan described in DCD Tier 2 Section 17 meets the requirements of 10 CFR 50.34(f)(3)(iii).

As discussed in Section 17.3 of this report, the NRC staff determined that the applicant maintained an NRC reviewed and approved QA program that complied with the requirements of 10 CFR 50, Appendix B. In addition, the staff plans to conduct an inspection of the implementation of the quality plan to verify that design activities conducted for the AP1000 project complied with the applicant's quality assurance program and the requirements of 10 CFR 50, Appendix B. The NRC staff plans to review the implementation of requirements related to the technically relevant portions of 10 CFR 50.34(f)(3)(iii) during this inspection. This issue is identified as Open Item 17.3-2.

10 CFR 52.47(a)(iv), requires, in part, that an application for design certification contain proposed technical resolutions of those medium- and high-priority Generic Safety Issues identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date six months prior to the application and which are technically relevant to the design. As discussed in NUREG-0933, the NRC staff resolved four issues associated with Item I.F.2 by establishing new requirements in SRP Chapter 17. These issues were:

- Item I.F.2(2) - Include QA personnel in review and approval of plant procedures
- Item I.F.2(3) - include QA personnel in all design, construction, installation, testing and operation activities

- Item I.F.2(6) - increase the size of the QA staff; and
- Item I.F.2(9) - clarify organizational reporting levels for the QA organization

The remainder of the issues associated with item I.F.2 were classified as low priority issues and therefore are not applicable to a design certification application. The staff concluded that, because Items I.F.2(2), (3), (6), and (9) were resolved by a revision to SRP Chapter 17, the NRC staff review of the quality assurance program conducted in accordance with the SRP Section 17.3 would verify compliance with these requirements. As discussed in Section 17.3 of this report, the staff determined that the applicant maintained an NRC reviewed and approved QA program that complied with the requirements for 10 CFR 50, Appendix B. In addition, the NRC staff plans to conduct an inspection of the implementation of the quality plan to verify that design activities conducted for the AP1000 project complied with the applicant's quality assurance program and the requirements of 10 CFR 50, Appendix B. The NRC staff plans to review the implementation of requirements related to the technically relevant portions of 10 CFR 50.34(f)(3)(iii) during this inspection. This issue is identified as Open Item 17.3-2.

I.G.1: Training Requirements

This TMI Action Plan Item called for a new operating license to conduct a set of low power tests to achieve the objectives of Task I.G, "Pre-operational and Low Power Testing." The objectives of Task I.G are to: (1) increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that the training for plant changes and off normal events was conducted; and (2) review the comprehensiveness of test programs. Near-term operating license facilities were required to develop and implement intensified training exercises during the low-power testing programs.

In Revision 3 to DCD Tier 2 Section 1.9.4, "AP1000 Resolution of Unresolved Safety Issues and Generic Safety Issues," the applicant addresses this item. The overall pre-operational and start-up testing program is addressed by the applicant in Chapter 14, "Initial Test Program. With regard to initial test program training, the applicant indicates, for example, that the results of performing natural circulation testing will be used as input into operator training. The applicant further states that data obtained from the first plant only natural circulation tests using SGs and passive residual heat removal (PRHR) is provided for operator training on the simulator at the earliest opportunity and, operator training for subsequent plants is also obtained while performing hot functional PRHR natural circulation testing.

The applicant identified that operator training program development is a COL applicant responsibility. In addition, Sections 14.2.5 of this report, Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program and 14.2.6, Trial Use of Plant Operating and Emergency Procedures, states that the NRC staff will defer the review of the trial use plant operating, emergency and testing procedures to the COL phase. The NRC staff agrees that operator training program development and implementation are the responsibilities of the COL applicant. Therefore, Issue I.G.1 is resolved for the AP1000 design.

Issue I.G.2: Scope of Test Program

10 CFR 52.47(a)(iv), requires, in part, that an application for design certification contain proposed technical resolutions of those medium- and high-priority Generic Safety Issues identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date six months prior to the application and which are technically relevant to the design. As discussed in NUREG-0933, the NRC staff resolved Item I.G.2, “Scope of Test Program,” by establishing new requirements in SRP Chapter 14.

In DCD Tier 2 Section 1.9.4, the applicant addressed this issue by stating that the program plan for preoperational and startup testing of the AP1000 design is in Section 14.2 which addressed Standard Review Plan Section 14.

The NRC staff has concluded that the initial test program conducted in accordance with SRP Section 14.2 is adequate. The staff evaluation of the test program, and the test program scope is documented in Section 14.2 of this report.

Issue II.B.1: Reactor Coolant System Vents

As discussed in NUREG-0933, Issue II.B.1 addressed the requirements in 10 CFR Part 50 and NUREG-0737 to install reactor vessel and RCS high-point vents. These vents are designed to release noncondensable gases from the RCS to avoid loss of core cooling during natural circulation. The design of these vents must conform to the following GDC requirements of 10 CFR Part 50 (Appendix A), and the applicable codes and standards for the RCS pressure boundary.

- the system must be operable from the control room (GDC 19)
- the system must be testable (GDC 36)
- the system must be capable of functioning following a LOOP (GDC 17)
- the system must be able to withstand an operating-basis earthquake (RG 1.29)

In DCD Tier 2 Section 1.9.3, item (2)(vi), the applicant states that in the AP1000 design the capability for remotely venting the high points of the RCS is provided by the safety-related Automatic Depressurization System (ADS) valves and the safety-related reactor vessel head vent system. Both discharge to the In-containment Refueling Water Storage Tank (IRWST). The ADS provides redundant groups of MOVs connected to the top of the pressurizer and air-operated valves connected to the top of each RCS hot leg. However, only the pressurizer MOVs, the first-stage ADS valves, are used for remote manual venting because they are the only ADS valves capable of being throttled. The reactor vessel head vent system removes steam and noncondensable gases directly from the reactor vessel head.

The applicant further states that, during normal and moderate frequency events, noncondensable gases from the RCS accumulate in the pressurizer steam space with very little

accumulation in the reactor vessel head, because of the continuous recirculation of bypass spray flow through the pressurizer when the RCPs are operating. This bypass flow causes boiling in the pressurizer, making the pressurizer steam space the lowest static pressure region in the RCS. This causes off-gassing of the RCS to occur in the pressurizer. This gas accumulation can be removed by remote manual operation of the first-stage ADS valves.

During LOCAs, the ADS automatically depressurizes the RCS so that the passive core cooling system may operate and effectively deliver cooling flow to the core. This would not happen until the RCS pressure was brought down to the passive core cooling system operating level.

The applicant also states that it is possible that continued depressurization of the RCS by the ADS could result in creation of a gas-steam volume (or bubble) in the upper region, or head, of the reactor vessel. With only the ADS, this volume can expand, filling the head of the vessel until it reaches the inside of the hot leg and is vented through the hot leg and the surge line, and out of the RCS. At the hot leg, this volume either vents into the pressurizer through the surge line and enters the ADS, or enters the ADS through the hot leg. This will depend on which ADS valves are open. This venting provides an open injection and steam venting path through the reactor vessel, and maintains required core flow without needing to refill the reactor vessel and pressurizer.

The staff reviewed the high-point vents for the AP1000 design. The design relies on the safety-related ADS valves and the safety-related reactor vessel head vent system to provide the capability of high-point venting of non-condensable gases from the RCS. DCD Tier 2 Sections 5.4.12 and 5.4.6 (and 6.3) provide descriptions of the reactor vessel head vent system and the ADS valves, respectively. These systems are operated from the control room, and associated valve position indications and alarms are provided. Their vent paths discharge to the IRWST.

The vessel head vent system is located entirely inside containment. Because the system isolation valves do not serve a containment isolation function, containment integrity will not be compromised as a result of a loss of power to the valves. This is a design improvement relative to current operating and standard design plants, where the reactor vessel head vent system isolation valves also provide containment isolation.

The system has the capability to remove noncondensable gases or steam from the RCS via remote manual operation of the redundant vent paths. It is designed to vent a volume of hydrogen equal to approximately 40 percent of the RCS volume at system pressure and temperature in 1 hour. The first-stage ADS valves are attached to the pressurizer and they provide the capability to vent noncondensable gases from the pressurizer steam space following an accident.

Sections 5.4.12 and 6.3 of this report provide the staff evaluation of the RVHV system and the ADS design, respectively. The staff concludes that the AP1000 design complies with Part 50.34(f)(2)(vi) requirements; therefore, Issue II.B.1 is resolved for the AP1000 design.

Issue II.B.2: Safety Review Consideration – Plant Shielding To Provide Post-accident Access to Vital Areas

As discussed in NUREG-0933, “A Prioritization of Generic Safety Issues,” Issue II.B.2 addressed having licensees perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The review would locate vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, where occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. This issue was resolved and the requirements were provided in 10 CFR 50.34(f)(2)(vii) (II.B.2).

In DCD Tier Section 2 1.9.3, Item (2)(vii), the applicant states that a plant shielding analysis was performed of the AP1000 general plant arrangement. This included a review of the primary shielding surrounding the reactor; the secondary shielding that encloses the reactor coolant loops; the shielding for refueling operations, including the refueling canal walls and refueling water; the auxiliary shielding such as equipment compartments, valve galleries, piping tunnels, the CVS, and other equipment modules; and accident shielding, including the shielding provided by buildings and the shielding to minimize sky shine. The applicant further states that improvements were incorporated into the AP1000 shielding design as they were identified.

DCD Tier 2 Section 12.2 addresses post-accident radiation sources used in the shield design and assessment of post-accident access to vital areas. The post-accident source term used for the AP1000 is predicated on the core release model from NUREG-1465, which supersedes the TID-14844 source term assumptions as reflected in RG 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors.” DCD Tier 2 Section 12.2 contains tables that list the post-LOCA instantaneous and integrated source strengths as a function of time. Section 12.3 addresses vital areas for post-accident access and includes radiation zone maps that show projected dose rates in these areas and access routes for the various post-accident actions requiring access to vital areas.

In DCD Tier 2 Section 12.4.1.8, the applicant provided a listing of the six vital plant areas that will require post-accident accessibility. For each of these areas, the applicant performed an analysis to determine the dose to the individuals performing these post-accident actions. These analyses, which utilized the appropriate time-dependent post-accident dose rates and the required post-accident access times, confirmed that personnel radiation doses for individuals accessing these areas following an accident will not exceed the guidelines of GDC 19 (5E-02 sieverts (5 rem) whole body or its equivalent to any part of the body) during the course of the accident.

The staff concludes that, on the bases of the information presented above, Issue II.B.2 is resolved for the AP1000 design.

Issue II.B.3: Postaccident Sampling Capability

The requirements for the postaccident sampling system are in 10 CFR 50.34(f)(2)(viii). The reactor coolant and containment atmosphere sampling-line systems should permit personnel to take a sample under accident conditions promptly and safely. The radiological spectrum analysis facilities should be capable of quantifying certain radionuclides that are indicators of the degree of core damage promptly. In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions.

The NRC published a model Safety Evaluation Report on eliminating the post-accident sampling system requirements from the TS for operating plants (Federal Register Volume 65, Number 211, October 31, 2000).

As discussed in NUREG-0933, Issue II.B.3 addressed upgrading postaccident sampling at plants. The requirements are in 10 CFR 50.34(f)(2)(viii). Issue II.B.3 specifically addressed a licensee's radiological and chemical sampling and analysis capability under transient or accident conditions, including related radiation exposures. The purpose of the postaccident Sampling System (PASS) is to provide sources of information for use by decision-makers in developing protective action recommendations, and to assess core damage.

In DCD Tier 2 Section 1.9.3, Item (2)(viii), the applicant states that the AP1000 sampling design-basis is consistent with the approach in the model safety evaluation and not the previous guidance of NUREG-0737 and RG 1.97. This guidance discusses contingency plans to obtain and analyze highly radioactive post-accident samples from the RCS and the containment sump and the containment atmosphere. The applicant states in DCD Tier 2 Section 1.9.3 that the AP1000 design is consistent with this guidance. Therefore, the staff finds the the applicant elimination of the postaccident sampling system for the AP1000 to be acceptable.

As discussed in Section 13.3.3.4 of this report, the PASS requirements of II.B.3 have been eliminated by the Model Safety Evaluation. The staff concludes that, based on the evaluation in Section 9.3.3 and 13.3 of this report, Issue II.B.3 is resolved for the AP1000 design.

Issue II.B.8: Rulemaking Proceedings on Degraded Core Accidents Description

Item II.B.8 of NUREG-0933 discussed the need to establish policy, goals, and requirements to address accidents resulting in core damage greater than the existing design basis. The Commission expects that new designs will achieve a higher standard of severe accident safety performance than previous designs. In an effort to provide this additional level of safety in the design of advanced nuclear power plants, the NRC developed guidance and goals for designers to strive for in accommodating events that are beyond what was previously known as the design-basis of the plant.

For advanced passive nuclear power plants, like the AP1000, the staff concluded that vendors should address severe accidents during the design stage to take full advantage of the insights gained from such input as probabilistic safety assessments, operating experience, severe accident research, and accident analysis by designing features to reduce the likelihood that

severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. Incorporating insights and design features during the design phase has been demonstrated to be much more cost effective than modifying existing plants.

The NRC issued guidance for addressing severe accidents in the following documents:

- the "NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants"
- the "NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants"
- the "NRC Policy Statement on Nuclear Power Plant Standardization"
- 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants"
- SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues And Their Relationship to Regulatory Requirements", and the corresponding staff requirements memorandum (SRM) dated June 26, 1990
- SECY-93-087, "Policy, Technical And Licensing Issues Pertaining to Evolutionary and Advanced Light-water Reactor (ALWR) Designs", and the corresponding SRM dated July 21, 1993

The first three documents provide guidance as to the appropriate course for addressing severe accidents, 10 CFR Part 52 contains general requirements for addressing severe accidents, and the SRMs relating to SECY-90-016 and SECY-93-087 give Commission-approved positions for implementing features in new designs for preventing severe accidents and mitigating their effects.

The basis for resolution of severe accident issues for the AP1000 is 10 CFR Part 52 and SECY-93-087. 10 CFR Part 52 requires (a) compliance with the TMI requirements in 10 CFR 50.34(f), (b) resolution of USIs and GSIs, and (c) completion of a design-specific probabilistic risk assessment. The staff evaluates these criteria in Sections 20.3, 20.1 and 20.2, and 19.1 of this report, respectively.

The Commission-approved positions on the issues discussed in SECY-93-087 form the basis for the staff's deterministic evaluation of severe accident performance for the AP1000. The staff evaluates the AP1000 relative to these criteria in Section 19.2 of this report. Issue II.B.8 is resolved for the AP1000 design on the basis of the staff's evaluation of the probabilistic and deterministic analyses in the AP1000 PRA, as documented in Chapter 19 of this report.

Issue II.D.1: Performance Testing of PWR Safety and Relief Valves

As discussed in NUREG-0933, Issue II.D.1 addressed the requirements in NUREG-0737 for qualification testing of RCS safety, relief, and block valves under expected operating conditions for design-basis transients and accidents, including ATWS. This issue was resolved by requiring licensees to conduct testing to qualify reactor coolant relief valves, safety valves, block valves, and associated discharge piping.

EPRI conducted a safety and relief valve test program for a group of PWR licensees to respond to the staff recommendations in NUREG-0587 and as clarified in NUREG-0737. The purpose of the program was to develop sufficient documentation and test data so that the participating licensees could demonstrate compliance with the II.D.1 requirements. The results were documented in the EPRI report, EPRI-NP-2770-LD, "EPRI PWR Safety Valve Test Report," December 1982. The staff used the test results documented in EPRI-NP-2770-LD and summarized in EPRI-NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Report," December 1982, as a part of its acceptance criteria in its evaluations of the resolution of Issue II.D.1 for all current operating plants.

In DCD Tier 2 Section 1.9.3, item (2)(x), the applicant states that the AP1000 design does not include PORVs and their associated block valves on the RCS. The safety valve and discharge piping used will either be of similar design as those valves tested and documented in EPRI Report NP-2770-LD or will be tested in accordance with the guidelines of Issue II.D.1 in NUREG-0737. The commitment in DCD Tier 2 Section 1.9.3 is consistent with the acceptance criteria used by the staff in it’s evaluations of Issue II.D.1 for operating plants, and is acceptable for the AP1000. Therefore, Issue II.D.1 is resolved for the AP1000 design.

Issue II.D.3: Coolant System Valves-Valve Posiion Indication

As discussed in NUREG-0933, Issue II.D.3 addressed the requirements in NUREG-0737 for positive indication in the control room of RCS relief or safety valve position. The acceptance criterion for the resolution of this issue is that the plant design shall include safety and relief valve indication derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe in accordance with the requirements in NUREG-0737.

This indication shall have the following design features:

- Unambiguous safety and relief valve indication shall be provided to the control room operator.
- Valve position should be indicated within the control room and should be alarmed.
- Valve position indication may be either safety or control grade; if it is control grade, it must be powered from a reliable (e.g., battery backed) instrument bus (see RG 1.97).
- Valve position indication should be seismically qualified consistent with the component or system to which it is attached.

- Valve position indication shall be qualified for the appropriate operating environment which includes the expected normal containment environment and an operating basis earthquake (OBE).

- Valve position indication shall be human-factors engineered.

In DCD Tier 2 Section 1.9.3 , item (2)(xi), the applicant states that the AP1000 design does not include PORVs and their associated block valves, and the direct indication of the position of the relief and safety valves in the AP1000 design is provided in the MCR.

This issue requires reactor coolant relief and safety valves be provided with positive indication in the control room. DCD Tier 2 Section 5.4.9 states that the AP1000 design complies with the requirements of 10 CFR 50.34(f)(2)(xi), positive position indication is provided for the pressurizer safety valves and the normal residual heat removal system relief valves. These valves are spring loaded, self-actuated by direct fluid pressure, and have back pressure compensation features. These valves are designed to re-close and prevent further flow of fluid after normal conditions have been restored. The pressurizer safety valves are of the totally enclosed pop type. The normal residual heat removal relief valve is designed for water relief.

Therefore, the staff considers that the AP1000 design satisfied the requirements of item II.D.3, and it is resolved for the AP1000 design.

Issue II.E.1.1: Auxiliary Feedwater System Evaluation

As discussed in NUREG-0933, Issue II.E.1.1 addressed improving the reliability of the auxiliary feedwater system or the emergency feedwater system. The issue addressed the following requirements in NUREG-0737:

- a simplified emergency feedwater (EFW) system reliability analysis to determine the potential for system failure under various loss-of-main-feedwater transients

- the acceptance criteria in SRP 10.4.9 and BTP ASB 10-1

- evaluated EFW flow rate design basis and criteria

In DCD Tier 2 Section 1.9.3, Items (1)(ii) and (2)(xii), the applicant states that the AP1000 design does not utilize an auxiliary feedwater system. A non-safety-related startup feedwater system (SUFWS) is provided to remove the core decay heat after the reactor trip during postulated non-LOCA events. Flow indication of the SUFWS is provided in the MCR. The SUFWS pumps automatically start following anticipated transients resulting in low SG level. The startup feedwater control valves automatically control feedwater flow to the SGs during operation. They can also be manually operated from the MCR. Operation of the SUFWS is not credited to mitigate licensing DBAs, which are discussed in DCD Tier 2 Chapter 15. The safety-related passive core cooling system provides emergency core decay heat removal during

transients, accidents, or whenever the normal non-safety-related heat removal paths are unavailable; it is described in DCD Tier 2 Section 6.3.

On the basis of the staff's review, which is discussed in Section 10.4.9 of this report, the staff concludes that Issue II.E.1.1 is resolved for the AP1000 design because the SUFWS is non-safety-related.

Issue II.E.1.2: AFW Automatic Initiation and Flow Indication

As discussed in NUREG-0933, Issue II.E.1.2 addressed improving the reliability of the auxiliary feedwater or emergency feedwater system. It addressed the requirement in NUREG-0737 for plants to install a control-grade system for automatic initiation of the EFWS. The acceptance criteria are in NUREG-0737 and in the design requirements of IEEE 279-1971. Specifically, the system shall incorporate such design features as automatic system initiation, protection from single-failure, and environmental and seismic equipment qualification. The issue requires provisions for automatic and manual auxiliary feedwater system initiation, and for flow indication in the control room.

In DCD Tier 2 Section 1.9.3, Items (1)(ii) and (2)(xii), the applicant states that the AP1000 design includes the non-safety-related startup feedwater system (SUFWS) and not an auxiliary feedwater system. Flow indication of the SUFWS is provided in the MCR. The SUFWS pumps automatically start following anticipated transients resulting in reactor trips and the control valves automatically control feedwater flow to the SGs during operation. They can also be manually operated from the MCR. The safety-related passive core cooling system provides emergency core decay heat removal during transients, accidents, or whenever the normal heat removal paths are unavailable.

The AP1000 design does not use an auxiliary feedwater system. The design uses a non-safety-related SUFWS to remove the core decay heat after a reactor trip during non-LOCA events. Because the SUFWS is non-safety-related and not taken credit for in an accident, the system does not have to meet all of the requirements of IEEE Standard 603-1991. However, flow indication is provided in the MCR, and the pumps automatically start following anticipated transients resulting in a reactor trip and automatically control feedwater flow to the SGs during power operation. They can also be manually operated from the MCR. The safety-related passive core cooling system provides for emergency core cooling during transients and accidents, where the normal heat removal paths are not available.

Although the AP1000 design does not have a safety-related auxiliary feedwater system, it provides the SUFWS, which adequately addresses the requirements in this issue.

Therefore, Issue II.E.1.2 is resolved for the AP1000 design.

Issue II.E.1.3: Update Standard Review Plan and Development of RG

As discussed in NUREG-0933, Issue II.E.1.3 addressed improving the reliability of the AFWS or the EFWS. Section 10.4.9 of the SRP was to be updated, and RG 1.26 was to be revised to

include these systems and possibly endorse certain standards. The SRP section was updated in July 1981; however, no additional public and occupational risk reduction was identified to support the need to revise the RG and it was not revised. This issue is resolved and the requirements were established in the changes to the SRP.

In DCD Tier 2 Section 1.9.4.2.1 the applicant states that this issue was a requirement to update SRP 10.4.9 to address the requirements of Item II.E.1.1 and Item II.F.1.2 for the AFWS. The SRP is written for the safety-related AFWS with a seismic Category 1 water source. A safety-related AFWS also functions as an EFWS to remove heat from the primary system when the main feedwater system is not available during emergency conditions. The AP1000 does not have an EFWS and does not include a seismic Category 1 water source for either the main or startup feedwater systems. The passive residual heat removal system provides the safety-related function to remove heat from the primary system when the main feedwater is not available. The design criteria for the SUFWS are predicated on operational and investment protection considerations and not the requirements of SRP 10.4.9 or RG 1.26.

The SUFWS does not have to meet the requirements of SRP 10.4.9 and RG 1.26 because of design difference. The SUFWS is a non-safety system and does not perform the safety function of an EFWS.

Therefore, the staff concludes that Issue II.E.1.3 is resolved for the AP1000 design.

Issue II.E.2.2: Research on Small-Break LOCAs and Anomalous Transients

As discussed in NUREG-0933, Issue II.E.2.2 addressed the NRC research programs focused on SBLOCAs and reactor transients. The programs included experimental research in the loss-of-flow tests (LOFT), Semiscale LOFT, B&W integral systems test facilities, systems engineering, and material effects programs, as well as analytical methods development and assessments in the code-development program.

The programs called for in this issue were completed by the NRC and showed that ECCSs will provide adequate core cooling for SBLOCAs and anomalous transients consistent with the single-failure criteria of Appendix K to 10 CFR Part 50. The application of the experimental data from the research programs to validate the conservatism of the licensing codes used in the SBLOCAs are addressed in Issue II.K.3(30) in this section.

The applicant did not address this issue in the DCD Tier 2. It concluded, in DCD Tier 2 Table 1.9-2, that this issue was not relevant to the AP1000 design because this issue was resolved with no new requirements.

The AP1000 design is a PWR with passive safety systems evolved from the AP600 design, which was the first passive Advanced Light-Water Reactor (ALWR) design reviewed by NRC. The distinguishing feature of these passive safety system design is a dependence on safety systems whose operation is driven by natural forces, such as gravity and stored mechanical energy.

For a design with passive safety systems and without a prototype plant that will be tested over an appropriate range of normal, transient, and accident conditions, the following requirements, are required by 10 CFR 52.47(b)(2)(i)(A):

- The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.

- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

- Sufficient data exist on the safety features to the design to assess the analytical tools used for safety and analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

The staff has considered how the research for the non-passive LWRs apply to the passive safety system design. While passive systems may be conceptually simpler than conventional active systems, they may be potentially more susceptible to system interactions that can upset the balance of forces upon which the passive systems depend on for their operation. It should be noted that these "passive" systems still rely on some active operation to place them in operation.

The applicant developed test programs design to investigate the passive reactor and containment safety systems, including component phenomenological (separate-effects) test, and integral-systems tests.

As described in Chapter 21 of this report, the staff has evaluated and concluded that applicant’s earlier test program conducted for AP600 is applicable to the AP1000 design, except for the liquid entrainment through the upper plenum, hot leg, and out the ADS stage-4 valves. The staff requested that the applicant provides additional test data with regard to liquid entrainment. In its letter of April 11, 2003, “Response to NRC Letter from J. E. Lyons to W. E. Cummins, ‘AP1000 Request for Data to Resolve Liquid Entrainment Requests for Additional Information,’ dated March 18, 2003,” DCP/NRC1572, the applicant committed to present new test data in support of AP1000 design certification. The applicant plans on using the APEX-1000 test facility at Oregon State University (OSU) to resolve the open issue regarding liquid entrainment.

The staff evaluation of the test programs in support of AP1000 is discussed in Chapter 21 of this report. The staff considers Issue II.E.2.2 resolved for the AP1000 design, except for the liquid entrainment issue.

Issue II.E.3.1, "Emergency Power Supply for Pressurizer Heaters"

Issue II.E.3.1 requires that emergency power be available to ensure that natural circulation can be maintained in the RCS if offsite power is lost, and pressurizer heater motive and control power shall interface with emergency buses through qualified devices.

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 containment refueling water storage tank water and to the passive containment cooling system without the pressurizer heaters. Pressurizer heaters are not required for safety and do not require power from the Class 1E system.

Therefore, Issue II.E.3.1 is resolved for the AP1000 design.

Issue II.E.4.1: Dedicated Hydrogen Penetrations

This issue remains open because DCD Tier 2 for the control of combustible gas in containment during accidents does not comply with current regulations.

The NRC has proposed major changes to 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,” and related changes to 10 CFR 50.34 and 10 CFR 52.47, along with the creation of a new rule, 10 CFR 50.46a (see 67 FR 50374, August 2, 2002). These proposed changes are meant to risk-inform the combustible gas control requirements, and constitute significant relaxations of the requirements. The staff plans to finalize the rule changes during 2003.

DCD Tier 2 is written in anticipation of these rule changes. As such, it is not in compliance with the current, more-restrictive regulations. Furthermore, until the proposed rule changes are final and effective, the staff cannot know for certain if the DCD will comply with the revised rule.

Therefore, the issue of containment combustible gas control, as well as Issue II.E.4.1, must remain open at this time.

This is an Open Item 6.2.5-1.

Issue II.E.4.2: Containment Isolation Dependability

As discussed in NUREG-0933, Issue II.E.4.2 addressed improving the reliability and capability of containment structures to reduce the radiological consequences to the public from accidents, including degraded core events. The issue specifically addressed the need for dependable isolation of containment penetrations.

In DCD Tier 2 Section 1.9.3, item (2)(xiv), the applicant states that the AP1000 containment isolation design satisfies NRC requirements, including post-TMI requirements. It further explains that two barriers are provided, one inside containment and one outside. These barriers are usually valves, but in some cases they are closed, seismic Category 1 piping

systems not connected to the RCS or to the containment atmosphere. The design incorporates a reduction in the number of containment penetrations compared to previous plant designs and the majority are normally closed. Those few that are normally open use "automatically closed," failed-close isolation valves. The penetrations do not automatically reopen on the resetting of the isolation signal. Containment isolation is automatically actuated by diverse signals, and can be manually initiated from the MCR. DCD Tier 2 Section 6.2.3 provides additional information.

The II.E.4.2 requirements are encompassed in the acceptance criteria for SRP 6.2.4. The staff therefore considered the relevant requirements in its review of the containment isolation system. Therefore, the staff concludes that Issue II.E.4.2 is resolved for the AP1000 design.

Issue II.E.4.4: Purging

Issue II.E.4.4 served to improve the vent/purge valve isolation reliability of pre-TMI facilities. The vent/purge isolation valve operators at many of those facilities were not originally selected with consideration of torque capability to close against LOCA dynamic forces. Also, II.E.4.4 restricted containment vent/purge operations to safety-related purposes, thus reducing the likelihood that the valves would be open in the event of a LOCA.

DCD Tier 2 states that the AP1000 will meet the II.E.4.4 requirement. DCD Tier 2 Section 6.2.3.1.3.F states that "Isolation valves are designed to have the capacity to close against the conditions that may exist during events requiring containment isolation." TSs will preclude unnecessary venting. Debris screens will be provided to protect the isolation valves from LOCA blowdown debris. Therefore, the staff concludes that Issue II.E.4.4 is resolved for the AP1000 design.

Issue II.E.5.1: Design Evaluation

As discussed in NUREG-0933, Issue II.E.5.1 addressed the requirement for B&W licensees to propose recommendations on hardware and procedural changes relative to the need for methods for damping primary system sensitivity to perturbations in the once-through SG. 10 CFR 50.34(f)(2)(xvi) states that a design criterion should be established for the allowable number of actuation cycles of the ECCS and RPS consistent with the expected occurrence rate of severe overcooling events, considering anticipated transients and accidents.

The applicant in DCD Tier 2 Section 1.9.3 provides a response to Issue II.E.5.1, stating that this issue applies only to B&W designs. 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. Operation of the ADS is not expected for either design-basis or best-estimate overcooling events. DCD Tier 2 Section 3.9.1 has additional information.

In the staff evaluation of Issue II.E.5.1 addressed in Section 3.9.1.1, “Design Transients,” of this report, the staff concludes that this issue is resolved for the AP1000 design.

Issue II.E.6.1: In Situ Testing of Valves, Test Adequacy Study

As discussed in NUREG-0933, Issue II.E.6.1 addressed the adequacy of the requirements for safety-related valve testing. Valve performance is critical to the successful functioning of a large number of plant safety systems. This issue was divided into the following four parts during resolution of the issue. As a result of this division, the resolution of Issue II.E.6.1 was subsumed by the resolution of the following:

(1) Testing of PIVs

(2) In-situ testing and surveillance of check valves

(3) Reevaluation of the thermal-overload protection provisions for MOVs in RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"

(4) Operability verification for MOVs in accordance with GL 89-10

The staff’s evaluations of the first two parts above, testing of PIVs and check valves, are discussed in Section 3.9.6.2 of this report. The last two parts in the list above were also addressed by the staff in Section 3.9.6.2 of this report as a part of the resolution of GL 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance.” Since the resolution of Issue II.E.6.1 was subsumed by the resolution of the four separate parts listed above, and based on the staff's evaluations and resolution of these issues as discussed above, the staff concludes that Issue II.E.6.1 is resolved for the AP1000 design.

Issue II.F.1: Additional Accident Monitoring Instrumentation

As discussed in NUREG-0933, Issue II.F.1 addressed providing instrumentation to monitor plant variables and systems during and following an accident. The issue addressed the need for plants to include instrumentation to measure, record, and read out in the control room the following containment parameters:

- pressure
- water level
- hydrogen concentration
- high-range radiation
- noble gas effluents

The staff clarified Issue II.F.1 in NUREG-0737 and requirements were issued. The radiation and noble gas effluent instrumentation is required to provide for continuous sampling of radioactive iodine and particulates at all potential accident release points, and for onsite capability to analyze and measure these samples. The acceptance criterion is the guidance in RG 1.97, "Instrumentation For Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 3, dated May 1983. NUREG-0660 also provides the requirements for a human factors analysis, which is to include the use of the indicators listed above by the operator during normal and abnormal plant conditions,

integration of these indicators in plant EOPs and operator training, the use of other alarms, and the need for prioritization of alarms.

DCD Tier 2 Section 1.9.3, Item (2)(xvii), the applicant states that the AP1000 post-accident monitoring system is described in DCD Tier 2 Chapter 7.

The system provides indication of the following plant parameters:

- Containment pressure
- Containment water level
- Containment radiation (high level)
- Noble gases effluents – to ascertain RCS integrity

In DCD Tier 2 Section 7.5, the applicant compares the AP1000 design against the criteria in Revision 3 of RG 1.97 and addresses accident monitoring instrumentation.

The hydrogen monitors are not part of post-accident monitoring. Other noble gas effluents are designated Type E variables and include information to permit the operator 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 Tier 2 Subsection 11.5.5 has additional discussions on measurement of radioactive effluents and conformance with RG 1.97. Section II.F.1(3) of NUREG-0737 requires that the reactor containment be equipped with two physically separate radiation monitoring systems that are capable of measuring up to 10^5 Gr/hr (10^7 R/hr) in the containment following an accident. In DCD Tier 2 Section 11.5.6.2, the applicant states that the AP1000 design will incorporate four electrically independent ion chambers located inside the containment to measure high-range gamma radiation. These detectors will be mounted on the inner containment wall in widely separated locations, and will have an unobstructed "view" of a representative volume of the containment atmosphere. The design and qualification of these monitors complies with the guidelines of RG 1.97 and 10 CFR 50.34(f)(2)(xvii), with respect to detector range, response, redundancy, separation, onsite calibration, and environmental design qualification. The staff, therefore, finds these monitors to be acceptable.

The AP1000 primary sampling system is designed to provide post-accident sampling functions (this was addressed in DCD Tier 2 Subsection 9.3.3.1). The human factors aspects of this issue are addressed in Chapter 18 of this report. Accident monitoring instrumentation is discussed in Section 7.5 of this report.

Therefore, the staff concludes that the AP1000 design meets the requirements of Issue II.F.1.

Issue II.F.2: Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

Title 10 CFR 50.34(f)(2)(xviii) requires that instruments be provided in the control room, which have an unambiguous indication of inadequate core cooling (ICC), 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. NUREG-0737, TMI Action Plan Item II.F.2, discusses the ICC phenomena and the need to have a reactor water level indication system that provides indication of reactor coolant void fraction when the RCPs are operating, and reactor vessel water level when the RCPs are tripped.

Before the TMI accident, an accepted operational practice of PWRs was to operate the RCPs, if they were available, during a LOCA to provide continued core cooling. During the TMI LOCA event with the stuck-open PORVs, the reactor coolant continued to leak through the open valves, the pressurizer level indicated high, and subsequent ICC occurred because the reactor coolant was highly voided. Nevertheless, core cooling was maintained with the continued operation of the RCPs. Subsequently, the RCPs were tripped, and because of high void content in the coolant, the water level dropped below the top of the core causing fuel damage. As a result of the TMI lessons learned, the reactor vessel water level indication system was added, specifically for PWRs, to ensure operator action to trip the RCPs following a LOCA, rather than later in the LOCA sequence to prevent an ICC event. NUREG/CR-5374, "Summary of Inadequate Core Cooling Instrumentation for U.S. Nuclear Power Plants," discusses acceptable approaches to instrumentation used to address ICC.

In DCD Tier 2 Subsection 1.9.3(2)(xviii), the applicant states that the AP1000 reactor system includes instrumentation for detecting voids in the reactor vessel head and other reactor vessel inventory deficits that could lead to ICC. The applicant also lists the AP1000 features that provide margin to or indication of ICC, with additional information provided in Tier 2 Subsections 6.3 and 7.5.

In response to the staff's RAI 440.127, the applicant explained that the AP1000 design concept is different from current operating plants in that the AP1000 design automatically trips the RCPs and initiates safeguard injections through the passive safety systems such as CMT, ADS, passive residual heat removal (PRHR) and IRWST to maintain core cooling in the event of a SBLOCA. It does not rely on a reactor vessel level indication system as do existing reactors, where reactor vessel level indication is important for operator actions to trip the RCPs, to monitor coolant mass in the vessel, and to manually depressurize the RCS in the event of ICC. There is no need in the AP1000 for the operator to trip the RCPs, to inject water into the core, or to manually depressurize the plant during a SBLOCA.

The instruments typically used in current PWRs include subcooling margin monitoring capability, core-exit thermocouples, and reactor vessel level indication system, which together would provide the operator with the ability to monitor the coolant conditions and to appropriately take actions to ensure core cooling during the approach to, and to recover from, the inadequate core cooling conditions. The AP1000 design includes subcooling margin monitoring capability, core-exit thermocouples, and the hot-leg level indication system. The AP1000 hot-leg level

indication system is different from the reactor vessel level indication systems currently used in the applicant plants.

The AP1000 hot-leg level indication is a safety-related level indication system, which consists of separate pressure taps that connect to the bottom of the hot leg, and to the top of the hot-leg bend leading to the SG. This system has the ability to provide indication of reactor water vessel level for a range spanning from the bottom of the hot leg to approximately the elevation of the vessel mating surface.

However, during the operation of the ADS to depressurize the plant, the reactor vessel water level will vary greatly and will not provide a reliable indication of ICC. The AP1000 hot-leg water level indication is not used to direct operator actions, even when the water level may potentially drop below the hot-leg level. Therefore, the water level is not an important indication for mitigation of ICC in the AP1000 design. The hot-leg level indication system is used, however, as a verification of reactor water inventory to terminate the recovery action in the ERGs for the ICC event.

Because the AP1000 design automatically trips the RCPs during a SBLOCA event, and because the operators are not prone to be misled by forced two-phase flow, the core exit temperature is an important and sufficient indication of an approach to ICC condition. The temperature reading provided by core-exit thermocouples has been appropriately included in the ERGs for plant recovery.

The staff reviewed the the applicant response and determined that for a SBLOCA event, a safeguard signal would automatically trip the RCPs, passive safety systems such as the CMT would automatically inject water into the core, the ADS would automatically initiate to depressurize the plant, the reactor coolant would automatically be cooled by the PRHR, and subsequent injection from the IRWST would occur. The staff also determined that for the AP1000 design, the core-exit thermocouples and the subcooling margin monitoring together would provide unambiguous indication of an approach to ICC, and the safety-related hot-leg level indication is only used to terminate the recovery action in the ERGs for the ICC event.

Therefore, the requirements for ICC, as discuss in 10 CFR 50.34(f)(2)(xviii), have been satisfied and Issue II.F.2 is resolved for the AP1000 design.

Issue II.F.3: Instrumentation for Monitoring Accident Conditions

As discussed in NUREG-0933, Issue II.F.3 addressed the adequacy and availability of instrumentation that monitors plant variables and systems during and following an accident that includes core damage. Before the TMI-2 accident, nuclear power generating stations were equipped with accident monitoring instrumentation using the guidance identified in RG 1.97 (Revision 1) and ANSI/ANS Standard 4.5, "Criteria for Accident Monitoring Functions in Light-Water Cooled Reactors."

The acceptance criterion for the resolution of this issue is that there shall be instrumentation of sufficient quantity, range, availability, and reliability to permit adequate monitoring of plant

variables and systems during and after an accident. Specifically, the instrumentation shall conform to the guidance in RG 1.97 (Revision 3) and ANSI/ANS Standard 4.5 and should provide sufficient information to the operator for (1) taking planned manual actions to shut the plant down safely; (2) determining whether the reactor trip, engineered-safety-feature systems, and manually initiated safety-related systems are performing their intended safety functions (i.e., reactivity control, core cooling, and maintaining RCS and containment integrity); and (3) determining the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, RCPB, and containment) and determining if a gross breach has occurred.

In DCD Tier 2 Section 1.9.3, Item (2)(xix), the applicant states that the AP1000 post-accident monitoring system is designed using RG 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 non-safety-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 Class 1E microprocessor-based system that provides instrumentation to monitor plant variables and systems during and following an accident. It consists of two independent, electrically-isolated, physically-separated divisions. Additional information is provided in the previous response for Issue II.F.1 above and in DCD Tier 2 Section 7.5.

Based on the review of the information provided, the staff concludes that the AP1000 design meets the requirements of Issue II.F.3.

Issue II.G.1: Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

As discussed in NUREG-0933, Issue II.G.1 addressed upgrading the emergency power for the pressurizer relief and block valves, and pressurizer level indicators. In accordance with the requirements in NUREG-0737, the pressurizer equipment must be supplied from an emergency source of power in the event of a LOOP.

In DCD Tier 2 Section 1.9.3, the applicant states that the AP1000 design does not include PORVs and their associated block valves. Pressurizer level indication is provided by instrumentation powered from the Class 1E dc power system. This system provides safety-related, uninterruptible power for Class 1E plant instrumentation, control, monitoring, and other vital functions, including safety-related components essential for safe shutdown of the plant. The system is designed such that these essential plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are not available.

Based on the review of the information provided, the staff concludes that the AP1000 design meets the requirements of Issue II.G.1.

Issue II.J.3.1: Organization and Staffing to Oversee Design and Construction

As discussed in NUREG-0933, Issue II.J.3.1 addressed requiring license applicants and licensees to improve the oversight of design, construction, and modification activities so that

they will gain the critical expertise necessary for the safe operation of the plant. This issue was included in Issue I.B.1.1, "Organization and Management Long Term Improvements," which was resolved with changes to RG 1.8, "Personnel Selection and Training," and RG 1.33, "Quality Assurance Program Requirements (Operation)."

In DCD Tier 2 Section 1.9.3, Item (3)(vii), the applicant states that a management plan has been developed for the AP1000 project that 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 AP1000 are to be provided by the COL applicant.

The organization for the plant beyond the AP1000 design, the construction of the plant, and the modification of the plant are outside the scope of design certification for the AP1000 design. A part of these concerns involves the organization of the COL applicant; however, the concerns regarding design of the plant outside of the AP1000 design and construction do not involve the organization of the site operation. Therefore, the COL applicant will have the responsibility for addressing these concerns as part of the COL licensing process. This is COL Action Item 20.4-3.

QA standards and the organization that the applicant used for the design of the AP1000 are discussed and found acceptable in Chapter 17 of this report. Furthermore, the applicant identifies in DCD Tier 2 Table 1.9-2 states that this item is the responsibility of the COL applicant.

Therefore, the staff concludes that Issue II.J.3.1 is resolved for the AP1000 design.

Issue II.J.4.1: Revise Deficiency Reporting Requirements

As discussed in NUREG-0933, Issue II.J.4.1 addressed assuring that all reportable items are reported promptly to the NRC and that the information submitted is complete. The issue was resolved when new requirements were issued in 10 CFR Part 21 and 10 CFR 50.55(e), on July 31, 1991 (56 FR 36091).

The COL applicant will have the responsibility for having the proper reporting procedures and addressing this issue as part of the licensing process. This is considered a part of the plant procedures development by the COL applicant. This is COL Action Item 20.4-4.

Also, as indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. Therefore, the staff concludes that Issue II.J.4.1 is resolved for the AP1000 design.

Issue II.K.1(3): Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents

As discussed in NUREG-0933, Issue II.K.1(3) requested licensees to have operating procedures for recognizing, preventing, and mitigating void formation in the RCS during transients and accidents to avoid loss of the core-cooling capability during natural circulation.

The staff has reviewed the resolution of Issue I.C.I and its related emergency response guideline ERG AES-0.2, “Natural Circulation Cooldown.” There are guidelines provided to direct the operators to cooldown and depressurize the plant using natural circulation conditions by dumping steam and subsequent RNS operation. These steps are specified to preclude any possible upper head voids formation and also to direct the operators to verify that a steam void does not exit in the vessel. The staff concludes that the ERGs provide directions to plant operators to recognize and to preclude voids formation in the vessel. Therefore, the staff considers Issue II.K.1(3) resolved for the AP1000 design.

Issue II.K.1(4d): Review Operating Procedures and Training to Ensure that Operators are Instructed Not to Rely on Level Alone in Evaluating Plant Conditions

As discussed in NUREG-0933, Issue II.K.1(4d) asked licensees to provide operating procedures to ensure that operators shall not rely on level indication alone in evaluating plant conditions. As stated in NUREG-0933, the staff determined that this issue was covered by Issues I.A.3.1, I.C.1, and II.F.2, and is resolved.

Issue I.A.3.1, "Revise Scope and Criteria for Licensing Examinations," was implemented by NRC by a rule change to 10 CFR Part 55, "Operators Licenses," to require a simulator as part of the reactor operator licensing examinations. The staff will impose the requirements of 10 CFR 55.45 on simulators on the COL applicant referencing the AP1000 design; therefore, the applicant and the staff does not have to address Issue I.A.3.1 for compliance with 10 CFR 52.47(a)(1)(iv).

The applicant does not address this issue in DCD Tier 2 information. It concludes, in Table 1.9-2, that this issue is not relevant to the AP1000 design because this issue is not a design certification issue, but is the responsibility of the COL applicant. However, the applicant has earlier stated that the design portion of this item is addressed in the proposed resolution to Issues I.C.1 and II.F.2.

The staff completed its review of Issues I.C.I and II.F.2 and concluded that AP1000 ERGs, as described in Section 18.9 of this report, do not instruct the operators to rely on level indication alone in evaluating plant conditions. The status of core cooling is determined by indications of core-exit thermocouple temperature, RCS subcooling, and RCS hotleg temperature in addition to RCS level. The staff considers these issues resolved. Therefore, Issue II.K.1(4d) is resolved for the AP1000 design.

Issue II.K.1(5): Safety-Related Valve Position Description

As discussed in NUREG-0933, Issue II.K.1(5) addressed the need to (1) review all valve positions and positioning requirements and positive controls, along with all related test and maintenance procedures to assure proper ESF functioning, if required, and (2) verify that auxiliary feedwater valves are in the open position. This issue was resolved and requirements were issued in NUREG-0737.

The applicant does not address this issue in the DCD Tier 2, and concluded, in Table 1.9-2, that this issue is not relevant to the AP1000 design because it is the responsibility of the COL applicant.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant has satisfactorily addressed this item in WCAP-14645. The staff has concluded that WCAP-14645 is applicable to AP1000 design, therefore, Issue II.K.1(5) is resolved.

Issue II.K.1(10): Review and Modify Procedures for Removing Safety-Related Systems From Service

As discussed in NUREG-0933, Issue II.K.1(10) addressed the requirement that licensees review and modify, as needed, the procedures for removing safety-related systems from service, and restoring them to service, to assure that the operability status of the systems is known.

The applicant did not initially address this issue in the DCD Tier 2, and concluded in Table 1.9-2, that this issue was not relevant to the AP1000 design because it was the responsibility of the COL applicant.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. The staff has concluded that WCAP-14645 is applicable to AP1000 design, therefore, Issue II.K.1(10) is resolved.

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

As discussed in NUREG-0933, Issue II.K.1(13) addressed the requirement for operating plants to propose TS reflecting the requirements in the bulletins issued by the Commission for the TMI Action Plan.

In DCD Tier 2 Chapter 1, Section 1.9.4.2.1, Westinghouse states that the AP1000 TS (DCD Tier 2 Section 16.1) are predicated on and were reviewed against the Westinghouse Standard TS (STS), which incorporated the requirements of the bulletins for the TMI Action Plan. The AP1000 TS are evaluated in Chapter 16 of this report. The staff reviewed the AP1000 TS against the STS, which incorporated all the requirements of the bulletins for the TMI Action Plan. Based on this review, the staff concludes that the approved AP1000 TS incorporate all the appropriate bulletin requirements from the TMI Action Plan. The incorporation of operating experience in bulletins in the AP1000 design is discussed in Section 20.7 of this report.

Therefore, Issue II.K.1(13) is resolved for the AP1000 design.

Issue II.K.1(16): Implemented Procedures that Identify Pressurizer PORV "Open" Indications and that Direct Operators to Close Valve Manually at "Reset" Setpoint

As discussed in NUREG-0933, Issue II.K.1(16) addressed requiring procedures that identified pressurizer PORV "open" indications and directed operators to close the valve manually at the "reset" setpoint. The staff determined in NUREG-0933 that this issue was covered by Issues I.C.1 and II.D.3.

In DCD Tier 2 Table 1.9-2, the applicant states that this issue is not applicable to the AP1000 design and the issue is only applicable to current operating plants. The staff agrees with the applicant assessment since this issue is related to PORV positions and AP1000 design does not include these valves. Therefore, Issue II.K.1(16) does not apply to the AP1000 design.

Issue II.K.1(17): Trip Pressurizer Level Bistable so that Pressurizer Low Pressure Will Initiate Safety Injection

As discussed in NUREG-0933, Item II.K.1(17) addresses the requirement for the applicant plants 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.

The AP1000 design does not depend on pressurizer low pressure and pressurizer low level coincidence to initiate safety injection in the event of LOCAs. Safety injection in AP1000 design is automatic. As described in DCD Tier 2 Subsection 7.3.1.1, the safeguard signals that initiate safety injection are Low-1 pressurizer pressure, Hi-2 containment pressure, low compensated steamline pressure, or Low cold-leg temperature. In addition, the AP1000 design also gives the operator manual safety injection capability. The staff concludes that any single safeguard signals mentioned above would initiate safety injection. Therefore, Issue II.K.1(17) is resolved for the AP1000 design.

Issue II.K.1(22): Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater is not Operable

As discussed in NUREG-0933, Issue II.K.1(22) addressed the requirement for BWR plants that auxiliary heat removal systems should be designed such that necessary automatic and manual actions can be taken to ensure proper functioning of the systems when the main feedwater system is not operable.

The applicant identifies in DCD Tier 2 Table 1.9-2 that it considers Issue II.K.1(22) relevant to the AP1000 design; however, this issue is not required for the AP1000 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In DCD Tier 2 Section 1.9.3, item (2)(xxi), the applicant states that, although this issue was applicable only to BWRs in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980, there are some considerations for the AP1000 design. Following a

loss of main feedwater (LMFW), there are a number of plant systems that automatically actuate to provide decay heat removal. The non-safety-related SUFWS can be powered by the non-safety-related diesel generators, and is automatically actuated and controlled by low SG level. For design-basis events, the safety-related passive core cooling system includes passive residual heat removal heat exchangers, which automatically actuate to provide emergency core decay heat removal if the non-safety-related systems are not available. The MCR meets the NRC guidelines for manual actuation of protective functions, including those used in a LMFW event. DCD Tier 2 Sections 6.3 and 10.4 provide additional information.

On the basis of the staff's review, which is discussed in 10.4.9 of this report, the staff concludes that Issue II.K.1(22) is resolved for the AP1000 design because the SUFWS is automatically actuated and controlled following a LMFW, and the passive core cooling system is automatically actuated if the non-safety related systems are not available.

Issue 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

Issue II.K.1(24) of NUREG-0933 required PWR licensees to perform a LOCA analysis for a range of small-break sizes and a range of time lapses between reactor trip and RCP trip. The staff determined in NUREG-0933 that this issue for PWRs was covered by Issue I.C.1, "Short-Term Analysis and Procedures Revision."

The staff has reviewed the responses to Issue I.C.1, and concluded that the AP1000 design automatically trips the RCPs during a LOCA event. The guideline directs the operators to verify that all RCPs have been tripped, and if not, the operators are directed to manually trip the RCPs. On the basis of the plant design features and the appropriate operator actions using the ERGs, the staff considered Issue II.K.1(24) resolved for the AP1000 design.

Issue II.K.1(25): Develop Operator Action Guidelines

As discussed in NUREG-0933, Issue II.K.1(25) required PWR licensees to develop operator action guidelines on the basis of the analyses performed in response to Issue II.K.1(24), which is discussed above. The staff determined in NUREG-0933 that this issue was covered by Issue I.C.1.

The applicant does not address this issue in DCD Tier 2, but indicated in Table 1.9-2 that this issue has been superseded by one or more other issues. Although this issue was covered by Issue I.C.1, as stated above, the applicant considered Issue I.C.1 the sole responsibility of the COL applicant.

The final procedures would be the responsibility of the COL applicant, however, the LOCA analyses for a range of time lapses and the specific information to go into the procedures would be the responsibility of the designer, or the applicant in the case of the AP1000 design. In DCD Tier 2 Chapter 15, the applicant addresses accidents for the AP1000 design.

As discussed earlier under section 20.4 of this report, the staff completed its review of Issue I.C.1 and concluded that Issue I.C.I is resolved. Therefore, Issue II.K.1(25) is resolved for the AP1000 design.

Issue II.K.1(26): Revise Emergency Procedures and Train Reactor Operators (ROs) and Senior Reactor Operators (SROs)

As discussed in NUREG-0933, Issue II.K.1(26) addressed requiring all operating PWRs to revise their EOPs and to train the ROs and SROs for the plant, on the basis of guidelines developed in response to Issue II.K.1(25), which is discussed above. The staff determined in NUREG-0933 that this issue is covered by Issues I.A.3.1, "Revise Scope of Criteria for Licensing Examinations," I.C.1, and I.G.1, "Training Requirements."

As stated in NUREG-0933, Issues I.A.3.1, I.C.1, and I.G.1 have been implemented in the staff review of reactor plant designs and do not have to be addressed by the applicant for compliance with 10 CFR 52.47(a)(1)(iv).

The applicant does not address this issue in the DCD Tier 2, but concludes, in Table 1.9-2, that this issue is not relevant to the AP1000 design because the issue has been superseded by one or more other issues. Although this issue was covered by Issues I.A.3.1, I.C.1, and I.G.1, as stated above, the applicant considered them the sole responsibility of the COL applicant.

Issue 1.C.1 was addressed earlier in this section. Issue I.A.3.1 was to revise the scope of examinations and criteria for licensing examinations and Issue I.G.1 was new training requirements for operators. In DCD Tier 2 Table 1.9-2, the applicant identifies Issues I.A.3.1 and I.G.1 as the responsibility of the COL applicant and not the responsibility of the applicant in the AP1000 design review.

The guidelines developed as part of Issue II.K.1(25) are discussed in the previous section. Also, as indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed Issue I.C.1 of this item in WCAP-14645 as staff has concluded that WCAP-14645 is applicable to AP1000 design.

However, the staff also identifies COL Action Item 20.4-5 for Issue II.K.1(26) that the COL applicant should address the scope of examinations and criteria for licensing examinations (Issue I.A.3.1), as well as new training requirements for operators (Issue I.G.1).

Therefore, Issue II.K.1(26) is resolved for the AP1000 design.

Issue II.K.1(27): Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling

As discussed in NUREG-0933, Issue II.K.1(27) addressed the need for PWR licensees to provide analyses and to develop guidelines and procedures for an inadequate core cooling (ICC) condition. The staff determined in NUREG-0933 that this issue was covered by Issues

I.C.1 and II.F.2. The resolution of Issues I.C.1 and II.F.2 for the AP1000 design were discussed earlier in this section.

The applicant does not address this issue in DCD Tier 2, but concluded in Table 1.9-2 that this issue is not relevant to the AP1000 design because the issue has been superseded by one or more other issues. Although this issue was covered by Issues I.C.1 and II.F.2, as stated above, the applicant considered this issue the sole responsibility of the COL applicant. The applicant addressed Issue II.F.2 in Tier 2 Subsection 1.9.3(2)(xviii).

As described in Section 18.9 of this report, the applicants ERGs provide high-level guidance to deal with inadequate core cooling conditions. The staff reviewed AFR-C.1, “AP600 Response to Inadequate Core Cooling Procedure and Analysis Bases,” which describes how passive safety-related systems would automatically trip the RCS pumps, and depressurize the RCS to inject water into the core upon receiving a safeguard signal. In this procedure, the operators are instructed to monitor plant conditions using core-exit temperature and indicated hot-leg level, which is designed to provide indication of an approach to ICC and to recover from an ICC condition. The operators are also instructed to manually initiate injection when automatic passive safety injections fail. Passive safety-related system actuation indications of CMT, ADS, PRHR, and IRWST are integrated into the procedures, which provide operators with directions to ensure that adequate core cooling will be maintained. The staff concluded that the above information is applicable to AP1000 design, as passive safety-related systems of AP600 are similar to the systems in AP1000.

The staff concludes that the applicant has appropriately provided analyses and procedures to mitigate ICC conditions. Therefore, Issue II.K.1(27) is resolved for the AP1000 design.

Issue II.K.1(28): Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required

As discussed in NUREG-0933, Issue II.K.1(28) addressed the requirement that PWRs be designed to ensure automatic RCP trip for all circumstances where required. The staff determined in NUREG-0933 that this issue was covered by Issue II.K.3(5), "Automatic Trip of RCPs During Loss-of-Coolant Accident."

The applicant does not address Issue II.K.1(28) in DCD Tier 2, but concluded, in Table 1.9-2, that this issue is not relevant to the AP1000 design because the issue had been superseded by one or more other issues, i.e., Issue II.K.3(5). In DCD Tier 2 Subsection 1.9.4.2.1, the applicant provides its response to Issue II.K.3(5), stating that the AP1000 design provides for an automatic trip of the RCPs on actuation of the passive core cooling system. This trip is provided to prevent RCP interaction with the operation of the core makeup tank. The staff concludes that Issue II.K.1(28), as well as II.K.3(5), is resolved for AP1000.

On the basis of the approved resolution of Issue II.K.3(5) for the AP1000 design, as discussed in this section, Issue II.K.1(28) is resolved for the AP1000 design.

Issue II.K.2(10): Hard-Wired Safety-Grade Anticipatory Reactor Trips

As discussed in NUREG-0933, Issue II.K.2(10) addressed the requirement for B&W plants to provide a design and schedule for implementation of a safety-grade reactor trip on loss of main feedwater, turbine trip, and significant reduction in SG level. These requirements were listed as Item 5 in BL 79-05B, which was issued on April 21, 1979. This issue was resolved and new requirements were issued.

In DCD Tier 2 Section 1.9.3, Item (2)(xxiii), the applicant states that this issue is applicable to only B&W plants, but that the AP1000 trip logic includes an anticipatory reactor trip for loss of main feedwater by using low SG water level. It also states that DCD Tier 2 Section 7.2 has additional information.

The applicant further states that, because the AP1000 design does not include PORVs and block valves, the anticipatory reactor trip on a turbine trip is not needed. The staff agrees with the the applicant statements and considers Issue II.K.2(10) resolved for the AP1000 design.

Issue II.K.2(16): Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power

As discussed in NUREG-0933, Issue II.K.2(16) required licensees to perform an evaluation of the likelihood and consequences of RCP seal damage following a small-break LOCA with a loss of offsite power.

In DCD Tier 2 Subsection 1.9.3, Item (1)(iii), the applicant states that the AP1000 design uses canned motor RCPs. This canned motor pump design does not have a seal that can fail and initiate RCS leakage.

The staff determined that this issue is covered by Issue 23, "RCP Seal Failures." The staff approved the resolution of Issue 23 for the AP1000 design in Section 20.3 of this report. Therefore, Issue II.K.2(16) is resolved for the AP1000 design.

Issue II.K.3(1): Install Automatic PORV Isolation System and Perform Operational Test

As discussed in NUREG-0933, Issue II.K.3(1) addressed, for PWR operating plants, the requirement in NUREG-0737 and NUREG-0660 to provide a system that uses the PORV block valves to protect against small-break LOCAs. This system would automatically cause the block valves to close when the RCS pressure decays after the PORV has opened.

In DCD Tier 2 Table 1.9.2 the applicant indicated that Issue II.K.3(1) is resolved for the AP1000 by establishment of new regulatory requirements and/or guidance. In DCD Tier 2 Section 1.9.3, Item (1)(iv), “Automatic Power-operated Relief Valve Isolation System (NUREG-0737 Item II.K.3.2),” the applicant states that the AP1000 design does not include PORVs. The pressurizer volume is about 40 percent larger than the pressurizer volume in current PWRs with a comparable power rating. This larger volume 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 RCS 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 to prevent exceeding the maximum RCS pressure limit.

It further states that 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 of when the valves open. The AP1000 design includes an automatic depressurization system (ADS) consisting of six parallel sets of two valves in series connected to the pressurizer and two parallel sets of two valves in series, with one set connected to each RCS hot-leg.

On the basis of this information and the fact that the AP1000 design does not have a PORV, Issue II.K.3(1) is resolved for the AP1000 design.

Issue II.K.3(2): Report on Overall Safety Effect of PORV Isolation System

As discussed in NUREG-0933, Issue II.K.3(2) required applicants to document the action to be taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV. The design purpose of PORVs is to prevent RCS overpressure and to reduce challenge to spring-operated safety valves for design-basis events.

In DCD Tier 2 Table 1.9.2 the applicant indicated that Issue II.K.3(1) is resolved for the AP1000 by establishment of new regulatory requirements and/or guidance. In DCD Tier 2 Section 1.9.3, Item (1)(iv), “Automatic Power-Operated Relief Valve Isolation System (NUREG-0737 Item II.K.3.2),” the applicant states that the AP1000 design does not include PORVs. It further describes the AP1000 pressurizer design, including the pressurizer safety valves and ADS to justify the appropriateness of the AP1000 design. (See the the discussion in Issue II.K.3(1) above.)

On the basis of the above discussion and the fact that the AP1000 design does not include a PORV, Issue II.K.3(2) is resolved for the AP1000 design.

Issue II.K.3(5): Automatic Trip of RCPs During LOCA

As discussed in NUREG-0933, Issue II.K.3(5) addressed requiring PWR licensees to study the need for an automatic trip of the RCPs and to modify plant procedures or the design, as appropriate. Licensees should know how to operate the RCPs in order to mitigate transients and accidents. The preservation of the maximum RCS inventory should be considered in the SBLOCA mitigation and the most effective decay heat removal (DHR) strategy should be considered in the mitigation of other transients.

In DCD Section 1.9.4.2.1, the applicant states that the AP1000 design provides for an automatic trip of the RCPs on actuation of the passive core cooling system. This trip is provided to prevent RCP interaction with the operation of the core makeup tank. Additional information regarding the automatic RCP trip is provided in DCD Tier 2 Section 6.3.

On the basis of this information, the staff concludes that Issue II.K.3(5) is resolved for the AP1000 design.

Issue II.K.3(6): Instrumentation to Verify Natural Circulation

As discussed in NUREG-0933, Issue II.K.3(6) addressed the requirement for licensees to provide instrumentation to verify natural circulation during transient conditions. The staff determined in NUREG-0933 that this issue was covered by Issues I.C.1, II.F.2, and II.F.3.

The applicant does not address this issue in Section 1.9.4.2.1 because DCD Tier 2 Table 1.9-2 indicates that this issue is superceded by other issues.

Based on the staff's review of the AP1000 design's compliance with TMI Action Items I.C.1, II.F.2 and II.F.3, described in the corresponding items in this report, the staff reached a conclusion that those issues relevant to the resolution of the TMI Action Item II.K.3(6) have been resolved. The detailed discussion of the related issues are addressed in their respective TMI item discussions. Therefore, Issue II.K.3(6) is resolved for the AP1000 design.

Issue II.K.3(8): Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of Steam Generators (SGs)

As discussed in NUREG-0933, Issue II.K.3(18) addresses further staff consideration of the need for diverse decay heat removal methods, which are independent of the SGs. The staff determined in NUREG-0933 that this issue was covered by Issues II.C.1, "Interim Reliability Evaluation Program," and II.E.3.3, "Coordinated Study of Shutdown Heat Removal Requirements." In NUREG-0933, the staff also stated that Issue II.E.3.3 was addressed in Issue A-45, "Shutdown Decay Heat Removal Requirements."

The applicant does not address this issue in Subsection 1.9.4.2.1, but indicated in DCD Tier 2 Table 1.9-2 that this issue is superceded by other issues. As stated in NUREG-0933, Issues A-45, II.C.1, and II.E.3.3 have been implemented in the staff review of reactor plant designs and do not have to be addressed by the applicant and the staff for compliance with 10 CFR 52.47(a)(1)(iv).

In DCD Tier 2 Appendix 19E, the applicant describes the AP1000 shutdown evaluation. Appendix 19E describes multiple decay heat removal capabilities independent of the SG. The detailed discussion of the multiple decay heat capabilities is included in Chapter 19.3 of this report. Based on the staff's review, Issue II.K.3(8) is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue II.K.3(9) addressed requiring the applicant plants to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs. The requirements were issued in NUREG-0737 and NUREG-0660.

The applicant identifies, in DCD Tier 2 Table 1.9-2, that it considers Issue II.K.3(9) is resolved by establishment of new regulatory requirements and/or guidance.

The applicant addresses this issue in DCD Tier 2 Section 1.9.4.2.1, item II.K.3(9) where it states that this issue is not applicable to the AP1000 design because the design does not have PORVs. Additional information is provided in DCD Tier 2 Sections 5.1.2 and 5.2.2 .

Based on the staff's review, Issue II.K.3(9) is resolved for the AP1000 design.

Issue II.K.3(18): Modification of ADS Logic – Feasibility Study and Modification for Increased Diversity for Some Event Sequences

As discussed in NUREG-0933, Issue II.K.3(18) addressed requiring BWR plants to modify the ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. A feasibility study and risk assessment study were required to determine the optimum approach. The requirements were issued in NUREG-0737 and NUREG-0660.

The applicant does not address this issue in Section 1.9.4.2.1, but indicated in DCD Tier 2 Table 1.9-2 that this issue is resolved by establishment of new regulatory requirements and/or guidance. In DCD Tier 2 Section 1.9.3, Item (1)(vii), the applicant states that although this issue is identified as applicable to BWRs only, it is applicable to the AP1000 design because the design uses an ADS with some similarity to that used on BWRs. The ADS automatically actuates on Low-1 CMT level, coincident with a CMT actuation signal. It is stated that manual actuation of the ADS is not required to maintain core cooling. As discussed in this section in Issue II.B.8 regarding degraded core accidents, the AP1000 PRA analysis confirms the use of the reliability of the automatic ADS actuation. The reliability of the automatic ADS actuation is incorporated throughout the PRA analysis and is evaluated by the staff in Section 19.1 of this report.

The actuation of ADS stages 2 and 3 occur on a set time delay after the actuation of the first stage, as discussed above. ADS stage 4 actuates on a Low-2 CMT level. Therefore, the staff agrees that manual actuation of the ADS is not required to maintain core cooling.

On the basis of the staff review of the ADS design discussed in Section 6.3 of this report, Issue II.K.3(18) is resolved for the AP1000 design.

Issue II.K.3(25): Effect of Loss of AC Power on Pump Seal

As discussed in NUREG-0933, Issue II.K.3(25) required licensees to determine, on a plant-specific basis, the consequences of a loss of cooling water to the RCP seal coolers. The

demonstrated adequacy of the seal design to withstand a loss of offsite power should prevent excessive loss of RCS inventory following an anticipated operational occurrence. The requirements for this issue in NUREG-0737 are that the consequences of a loss of cooling water to the pump seal coolers be determined, and that the pump seals should be designed to withstand a complete LOOP for at least 2 hours.

The applicant does not address this issue in Section 1.9.4.2.1, but indicated that DCD Tier 2 Table 1.9-2 that this issue is resolved by establishment of new regulatory requirements and/or guidance. In DCD Tier 2 Section 1.9.3, Item (1)(iii), the applicant states that the AP1000 design uses canned motor RCP pumps. The canned motor pump design does not have a seal that can fail and initiate RCS leakage.

The staff determined that this issue was covered by Issue 23, "RCP Seal Failures," which is discussed in Section 20.3 of this report. On the basis of the approved resolution of Issue 23 for the AP1000 design in Section 20.3 of this report, Issue II.K.3(25) is resolved for the AP1000 design.

Issue II.K.3(28): Study and Verify Qualification of Accumulators on ADS Valves

As discussed in NUREG-0933, Issue II.K.3(28) addressed requiring assurance from BWR licensees that air or nitrogen accumulators for ADS valves had sufficient capacity to cycle the valves open five times at design pressure. The requirements were issued in NUREG-0737 and NUREG-0660.

The applicant does not address this issue in Section 1.9.4.2.1, but indicated that DCD Tier 2 Table 1.9-2 that this issue is resolved by establishment of new regulatory requirements and/or guidance. In DCD Tier 2 Section 1.9.3, Item (1)(x), the applicant states that 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 ADS 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 non-safety-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 to pre-operational and in-service testing. They will be included in the reliability assurance program. Based on the staff's review, Issue II.K.3(28) is resolved for the AP1000 design.

Issue II.K.3(30): Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix K

As discussed in NUREG-0933, Issue II.K.3(30) required licensees to revise and submit analytical methods for small-break LOCA analyses for compliance with Appendix K to 10 CFR Part 50 for NRC review and approval. The revised LOCA methods were to account for comparisons with experimental data, including data from LOFT test and Semiscale test facilities. Alternatively, licensees were to provide additional justification for the acceptability of

their SBLOCA models with LOFT and Semiscale test data. Clarifications were issued in NUREG-0737.

DCD Tier 2 Table 1.9-2 indicates that this issue is not an AP1000 design certification issue because the issue is applicable to current operating plants or responsibility of combined license applicant.

The AP1000 small-break LOCA analysis is performed with the NOTRUMP computer code. NOTRUMP was developed by the applicant to better address the thermal-hydraulic aspects of SBLOCA, which had become an issue following the accident at TMI. The staff reviewed the applicant NOTRUMP code including comparisons with experimental data, and documented its discussions in Chapters 15 and 21 of this report. Based on the staff's review, Issue II.K.3(30) is resolved for the AP1000 design.

Issue III.A.1.2: Upgrade Emergency Support Facilities

As discussed in NUREG-0933, Issue III.A.1.2 addressed requiring licensees to upgrade their emergency support facilities by establishing a technical support center (TSC), an operational support center (OSC), and a nearsite emergency operations facility (EOF) for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. This issue was resolved and new requirements were issued in NUREG-0737 ("Clarification of TMI Action Plan Requirements") and Supplement 1 to NUREG-0737 ("Requirements for Emergency Response Capability," GL No. 82-33, December 17, 1982).

The applicant identifies, in DCD Tier 2 Table 1.9-2, that it considers Issue III.A.1.2 relevant to the AP1000 design. In DCD Tier 2 Section 1.9.3, Item (2)(xxv) the applicant states that the AP1000 design provides for an onsite TSC and onsite OSC, which are discussed in DCD Tier 2 Chapter 18, and that the offsite emergency response facility is the responsibility of the COL applicant. The mission, major tasks, and location of the TSC and OSC for the AP1000 standard design are provided in DCD Tier 2 Sections 18.8.3.5 and 18.8.3.6, respectively. Figures 1.2-19 and 1.2-18 show the location of the TSC and OSC, respectively. The applicant states in DCD Tier 2 Section 13.3 that emergency planning is the responsibility of the COL applicant. This is reflected in Section 13.3.2 of this report as COL Action Item 13.3-1.

Additionally as discussed in section 13.3.3 of this report, certain design features, facilities, functions and equipment necessary for emergency planning must be considered in the standard design. Specifically, in accordance with 10 CFR 50.34(f)(2)(xxv), the standard design must address the characteristics of the onsite TSC and OSC. Section 13.3.3 of this report indicates that there are Open Items associated with the AP1000 emergency support facilities. Therefore, the staff concludes that, based on the evaluation in section 13.3.3 of this report, and specifically the identified Open Items, Issue III.A.1.2, is not resolved for the AP1000 design.

Issue III.A.3.3: Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Radio Communication Systems

As discussed in NUREG-0933, Issue III.A.3.3 addressed the need for licensees to upgrade their communications capability at the emergency support facilities at the plant listed in Issue III.A.1.2. Relevant guidance is contained in NUREG-0660.

DCD Tier 2 Section 13.3 states that emergency planning, including communication interfaces among the main control room, the technical support center and the emergency planning centers, are the responsibility of the COL applicant. Further, the COL applicant referencing the AP1000 certified design will address emergency planning, including post 72-hour actions and its communications interface. DCD Tier 2 Section 9.5.2 provides that COL applicants referencing the AP1000 certified design will address interfaces to required offsite locations. Further, the emergency response facility communication system, including the crisis management radio system, will be addressed by the COL applicant.

The staff considers that this issue is outside the scope of the AP1000 design certification, and that it, therefore, will be addressed by the COL applicant. This issue is covered by DCD Tier 2 Section 13.3, and is addressed in Section 13.3.2 of this report as COL Action Item 13.3.-1. Therefore, Issue III.A.3.3 is resolved for the AP1000 design.

Issue III.D.1.1: Primary Coolant Sources Outside the Containment Structure

As discussed in NUREG-0933, Issue III.D.1.1 addressed the requirement that licensees identify design features to reduce the potential for exposure to workers at plants and to offsite populations from the release of primary coolant following an accident. This issue has three subissues:

- III.D.1.1(1), "Review Information Submitted by Licensees Pertaining to Reducing Leakage From Operating Plants"

- III.D.1.1(2), "Review Information on Provisions for Leak Detection"

- III.D.1.1(3), "Develop Proposed System Acceptance Criteria"

In NUREG-0737, Subissue III.D.1.1(1) required licensees to implement a program to reduce leakage from systems outside the containment that would or could contain highly radioactive fluids during a serious transient, or following an accident, to as-low-as-practical levels.

For Subissue III.D.1.1(2), the staff also stated in NUREG-0933 that Issue II.F.1 addressed accident monitoring instrumentation and that the RCPB leak detection capability must be equivalent to that specified in RG 1.45. Issue II.F.1 is addressed for the AP1000 design in this section.

The need for requiring leak-detection systems and the development of new acceptance criteria for these systems in Subissue III.D.1.1(3) were pursued by the staff in other issues, as

Subissue III.D.1.1(2). Therefore, work on Subissue III.D.1.1(3) did not provide any data for staff consideration and this issue was dropped from further consideration.

In DCD Tier 2 Section 1.9.3, Item (2)(xxvi), the applicant states that the safety-related passive systems for the AP1000 design do not recirculate radioactive fluids outside the containment following an accident. A non-safety-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. Based on the staff's review, Issue III.D.1.1 is resolved for the AP1000 design.

Issue III.D.3.3: In-Plant Radiation Monitoring

10 CFR 50.34(f)(2)(xxvii) (III.D.3.3) states that the licensee shall "provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

The AP1000 will be provided with area and airborne radiation monitors to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in DCD Tier 2 Section 12.5. These area and airborne radiation monitors, which are described in DCD Tier 2 Section 11.5, will comply with the personnel radiation protection guidelines of 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 70, and RGs 1.97, 8.2, 8.8, and 8.12.

In addition, NUREG-0737, Item III.D.3.3 states that "each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident." Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments. Because the description of portable instrumentation, training, and procedures is outside the scope of DCD Tier 2, the applicant addressed this as a COL item.

In DCD Tier 2 Section 12.3.5, the applicant states that the COL applicant will address the criteria and methods for obtaining representative measurements of radiological conditions, including airborne radioactivity in work areas. In addition, the COL applicant will address the use of portable instruments and the associated training and procedures to accurately determine the airborne iodine concentrations in areas within the facility where plant personnel may be present during an accident.

The information on in-plant radiation monitoring in Chapter 12 of DCD Tier 2 addresses the requirements of 10 CFR 50.34(f)(2)(xxvii) (III.D.3.3) and the staff's concerns in this area are resolved. Therefore, Issue III.D.3.3 is resolved for the AP1000 design.

Issue III.D.3.4: Control Room Habitability

As discussed in NUREG-0933, Issue III.D.3.4 addressed upgrading the habitability of the control room for the operators. The requirements were provided in NUREG-0737.

In DCD Tier 2 Section 1.9.3, item (2)(xxviii), the applicant states that normally a non-safety-related HVAC system keeps the AP1000 MCR slightly pressurized to prevent infiltration of air from other plant areas. During accident conditions, safety-related isolation of the MCR is automatically actuated. Upon the loss of non-safety related ac power, the MCR 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 DBA conditions.

The MCR is stated to be 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. The onsite non-safety related normal HVAC system will be operational before the installed compressed air supply is exhausted.

It is further stated that the non-safety-related HVAC system, equipped with a refrigeration-type air conditioning unit and powered from the onsite diesel generators, normally provides MCR cooling. If the normal HVAC system is not available, outside air is not allowed into the MCR, and the safety-related compressed air storage system is actuated.

In the DSER, the applicant addressed the possibility of toxic gases and substances onsite and offsite affecting control room habitability; the signals, or procedures and operator action for actuation of equipment for control room habitability, are the responsibility of the COL applicant. DCD Tier 2 Section 6.4.7 states that the COL applicant referencing the AP1000 certified design is responsible for the amount and location of possible sources of toxic chemicals in or near the plant and provision for seismic Category 1, Class 1E toxic gas monitoring.

The applicant committed to conform with the guidance of RG 1.78-December 2001, Revision 1, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release” to meet the requirements of TMI Action Plan Item IIID.3.4 and GDC 19. In addition, the applicant will identify RG 1.78-December 2001, Revision 1, for DCD Tier 2 Sections 6.4.8, 9.4.13, DCD Tier 2 Table 1.9-1, and DCD Tier 2 Appendix 1A. The NRC Staff expects that Revision 4 of the DCD will reflect DCD Tier 2 Sections 6.4.8, 9.4.13, DCD Tier 2 Appendix 1A, and DCD Tier 2 Chapter 16, B3.7.6 accordingly. Therefore, this is Confirmatory Item 6.4-1.

The applicant submitted the results of radiological consequence analyses for personnel in the MCR during a DBA in DCD Tier 2 Section 6.4.4. Details of the analysis assumptions for modeling the doses to MCR personnel were submitted in DCD Tier 2 Section 15.6.5.3. The staff’s review and independent dose assessment will be completed once questions on the assumed aerosol removal rates in the containment, as discussed in unresolved RAIs 470.009 and 470.011, have been resolved. This issue is identified as An Open Item 6.4-1. Therefore, Issue III.D.3.4 is unresolved for the AP1000 design.

20.5 Human Factors Issues

Issue HF1.1: Shift Staffing

This issue addressed ensuring that the numbers and capabilities of the staff at nuclear power plants are adequate to operate the plant safely. This issue was to determine the minimum appropriate shift crew staffing composition. To meet this goal, consideration was given to the following: the number and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operational mode; the minimum qualifications of plant personnel in terms of education, skill, knowledge, training experience, and fitness for duty; appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation.

The review criteria for this issue are contained in the 10 CFR 50.54, SRP Sections 13.1.2 through 13.1.3, and RG 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit." The applicant does not address this issue in the DCD Tier 2. It concludes, in Table 1.9-2, that this issue is not relevant to the AP1000 design because this issue is the responsibility of the COL applicant. As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. Therefore, Issue HF1.1 is resolved for the AP1000 design.

Issue HF4.1: Inspection Procedure for Upgraded Emergency Operating Procedures

As discussed in NUREG-0933, Issue HF4.1 addressed the development of criteria by the NRC to provide assurance during inspections that operating plant EOPs are adequate and can be used effectively. Lessons learned by the staff from its inspections of EOPs at plants were published in NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures," April 1989. Temporary Instruction (TI) 2515/92, "Emergency Operating Procedures Team Inspections," was later issued containing guidance for conducting these inspections.

The issue was resolved with no new requirements. In DCD Tier 2 Section 1.9.4, the applicant states that the design of the AP1000 EOPs is consistent with NUREG-1358 and its supplements, as well as current regulatory guidance and standards. DCD Tier 2 Section 18.9.8 has additional information. This issue is covered in DCD Tier 2 Section 18.9, “Procedure Development,” and is a COL responsibility. Therefore, Issue HF4.1 is resolved for the AP1000 design.

Issue HF4.4: Guidelines for Upgrading Other Procedures

As discussed in NUREG-0933, this issue addresses efforts by the staff to evaluate the quality of, and the problems associated with, existing plant procedures to ensure that plant procedures (other than EOPs which are discussed in Issue HF4.1 above) are adequate and could be used effectively, and to guide operators in maintaining plants in a safe state under all operating conditions. The NRC was to evaluate the need to develop technical guidance for use by industry in upgrading normal operating procedures and abnormal operating procedures. The

objective of this issue is to be satisfied by (1) developing guidelines for preparing and criteria for evaluating normal operating procedures and other procedures that affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

The review criteria for this issue are in SRP Sections 13.5.1, "Administration Procedures," and 13.5.2, "Operating and Maintenance Procedures," and in Information Notice 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures." In addition, this item is covered by Element 8, Procedures Development, of NUREG-0711.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645, as the staff has concluded that WCAP-14645 is applicable to AP1000 design. Based on the staff's review, Issue HF4.4 is resolved for the AP1000 design.

Issue HF5.1: Man-Machine Interface – Local Control Stations

As discussed in NUREG-0933, Issue HF5.1 addressed assuring that the man-machine interface at local control stations and auxiliary operator interfaces is adequate for the safe operation and maintenance of a nuclear power plant. The concerns associated with this issue include the assurance that indications and controls made available to operators at local control stations outside of the control room and remote shutdown room are sufficient and appropriate for their intended use. The regulatory guidance has been limited to the control room and the remote shutdown panel. Control room crew activities should be analyzed to establish and describe communication and control links between the control room and the auxiliary control stations. Additionally, the potential impact of auxiliary personnel on plant safety should be analyzed.

This issue was resolved and no new requirements were established. In DCD Tier 2 Section 1.9.4, the applicant states that it uses techniques and experience gained in the design of the MCR and remote shutdown panel on the local control station panels. The methodology to analyze the job/tasks of the control room is stated to be applied to the job/tasks of auxiliary personnel to identify and describe communication and action links between the control room and the auxiliary control stations. As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. Therefore, Issue HF5.1 is resolved for the AP1000 design.

Issue HF5.2: Review Criteria for Human Factors Aspects of Advance Controls and Instrumentation

As discussed in NUREG-0933, Issue HF5.2 addressed the use of advanced I&Cs, in particular with respect to plant annunciators. The then-existing human engineering guidelines for control rooms addressed the control, display, and information concepts and technologies that were being used in process control systems. These guidelines were not believed to be adequate for advanced and developing technologies that could be introduced into future designs. Improved alarm systems using advanced technologies were expected to become available, and

guidelines for the use and evaluation of these longer-term alarm improvements were to be developed.

This issue focused on the potential risk that could result from the human error in the use of control room alarms. Work on this issue was stopped when the development of review guidance for advanced alarms was integrated into the Office of Nuclear Regulatory Research (RES) program to develop an "Advanced Human-Interface Design Review Guideline."

This issue is resolved with no new requirements. In DCD Tier 2 Section 1.9.4, the applicant states that the AP1000 advanced alarm design, described in DCD Tier 2 Section 18.9.2, conforms with current guidance and requirements on integrated human factors design. A description of the computerized procedures is in DCD Tier 2 Section 18.9.8.6. A detailed description of the qualified display processing system is in DCD Tier 2 Section 18.9.5. The plan for the V&V of the AP1000 man-machine interface system (M-MIS) is in DCD Tier 2 Section 18.8.2.3

The applicant identified and discussed the "current guidance and requirements on integrated human factors design" it used to design the advanced alarm system for the AP1000 design. The relationship of the computerized procedures and qualified display processing system to the advanced alarm system was also explained. As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, the applicant satisfactorily addressed this item in WCAP-14645. Therefore, Issue HF5.2 is resolved for the AP1000 design.

20.6 Three Mile Island Action Plan Requirements

Pursuant to 10 CFR 52.47(a)(ii), an applicant for design certification must demonstrate compliance with any technically relevant TMI requirements in 10 CFR 50.34(f). The relevant TMI Action Plan items, the 10 CFR 50.34(f) requirements, and the section in which they are addressed are listed in Table 20.6-1 of this report.

Table 20.6-1 10 CFR 52.47(a)(1)(ii) TMI Action Plan Items

TMI REQUIREMENT	10 CFR 50.34(f)	DSER Chapter/Section
I.A.4.2	(2)(i)	20.4
I.C.5	(3)(i)	20.4
I.C.9	(2)(ii)	20.4
I.D.1	(2)(iii)	18, 20.4
I.D.2	(2)(iv)	18, 20.4
I.D.3	(2)(v)	20.4

TMI REQUIREMENT	10 CFR 50.34(f)	DSER Chapter/Section
I.F.1	(3)(ii)	17, 20.4
I.F.2	(3)(iii)	17, 20.4
II.B.1	(2)(vi)	20.4
II.B.2	(2)(vii)	12, 20.4
II.B.3	(2)(viii)	9, 13, 20.4
II.B.8	(1)(i) & (xii), (2)(ix), (3)(iv) & (v)	19, 19, 20.4
II.D.1	(2)(x)	20.4
II.D.3	(2)(xi)	20.4
II.E.1.1	(1)(ii)	10, 19, 20.4
II.E.1.2	(1)(ii), (2)(xii)	20.4
II.E.3.1	(2)(xiii)	8, 20.4
II.E.4.1	(3)(vi)	20.4
II.E.4.2	(2)(xiv)	20.4
II.E.4.4	(2)(xv)	20.4
II.F.1	(2)(xvii)	7, 11, 20.4
II.F.2	(2)(xviii)	20.4
II.F.3	(2)(xix)	20.4
II.G.1	(2)(xx)	8, 20.4
II.J.3.1	(3)(vii)	20.4
II.K.2(16)*	(1)(iii)	20.4
II.K.3(2)	(1)(iv)	20.4
II.K.3(18)*	(1)(vii)	20.4
II.K.3(25)*	(1)(iii)	20.4
II.K.3(28)*	(1)(x)	20.4
III.A.1.2	(2)(xxv)	13, 18, 20.4
III.D.1.1	(2)(xxvi)	20.4

TMI REQUIREMENT	10 CFR 50.34(f)	DSER Chapter/Section
III.D.3.3	(2)(xxvii)	12, 20.4
III.D.3.4	(2)(xxviii)	6, 20.4

*Although these TMI Action Plan items did not apply to Westinghouse PWRs in NUREG-0737, they are applied to all PWR designs in 10 CFR 50.34(f)(1)(iii).

20.7 Incorporation of Operating Experience

20.7.1 Background

As part of its program to disseminate information on operational reactor experience to the nuclear industry, the NRC issues generic communications (bulletins, GLs, and information notices) when a significant safety-related event or condition at one or more facilities is believed to potentially apply to other facilities. A bulletin or GL is typically issued when the NRC staff determines that licensees should be required to inform the NRC what actions have been or will be taken to address an event, condition, or circumstance that is both potentially safety-significant and generic. An information notice is typically issued when the NRC staff determines that licensees should be informed of an event, condition, or circumstance that may be both potentially safety-significant and generic, but the event, condition, or circumstance is not sufficiently significant to warrant requiring licensees to confirm in writing that actions have been or will be taken. Potential safety issues highlighted in NRC generic communications have resulted in the establishment of a USI or GSI, and have also been incorporated into formal regulatory requirements.

The Commission requested, in its SRMs dated July 31, 1989, and February 15, and March 5, 1991, that an applicant submitting plant designs for standard plant design certification provide a discussion of how operating experience has been incorporated into the design.

A review of the AP1000 design for incorporation of important lessons learned from operating plant experience was accomplished by reviewing the bulletins and GLs issued between January 1, 1980, and December 31, 2002, and determining whether the applicant properly incorporated in the AP1000 design the staff positions in those documents that were applicable to the design. In the NRC programs that account for operating experience, the bulletins and GLs issued to the nuclear industry, convey the most safety-significant lessons distilled from numerous sources of information on operating plant malfunctions (e.g., Licensee Event Reports), issue staff positions on resolving problems in these malfunctions, and request actions to be taken by the licensees. As a contrast, information notices do not request actions on the part of the licensees. Thus, reviewing how applicable bulletins and GLs have been incorporated into the AP1000 design is a sufficient basis for reviewing the design against operating experience.

In the resolution of bulletins and GLs for the AP1000 design, the staff went outside the specific purpose of the documents to determine their resolution for the AP1000 design. In addition, some of the bulletins and GLs involved issues that will be the responsibility of the COL applicant during the construction or operation of the plant. This will be identified in the resolution of these documents for the AP1000 design.

20.7.2 Application Content Review

The applicant submitted DCD Tier 2, for standard plant design certification of the AP1000 design. In that document, the applicant states that the design engineers continually review industry experience from sources such as NRC bulletins, Licensee Event Reports, NRC requests for information, and GLs. It further states that operating plant experience has been incorporated in the AP1000 design by virtue of its participation in developing Volume III of the EPRI ALWR Utility Requirements Document and in the activities of the ALWR Utility Steering Committee.

The applicant also submitted WCAP-15800, "Operational Assessment for AP1000," to address the manner in which it incorporated operating plant experience into the AP1000 design and stated that it reviewed the NRC bulletins, GLs, circulars, information notices, and Office of Analysis and Evaluation of Operational Data (AEOD) reports for the time period January 1, 1980 to December 31, 2002. The applicant discussed the applicability of these NRC documents to the AP1000 design by referring to the appropriate DCD Tier 2 sections or explaining the AP1000 design does not have the equipment discussed in the NRC document. The disposition of the individual documents were broken down into the following categories:

- not applicable to the AP1000 design (e.g., BWR only, B&W or CE facilities only, or not applicable to commercial reactors)

- not applicable for other reason (e.g., procurement issue, administrative communication, procedural issue, maintenance or surveillance issue, plant specific or isolated event)

- applicable to AP1000 design certification

The staff considered the bulletins and GLs the applicant concluded were applicable to the AP1000 design in determining the list of such documents that the staff should review for how operating experience was incorporated in the design.

20.7.2.1 Regulatory Review

The SRP (NUREG-0800) guides the NRC staff for its review of a reactor facility design. This document states requirements, acceptance criteria (some of which are predicated on operating reactor experience), and findings that the staff must make. This document was last revised in April 1982. Significant issues raised before January 1981, were incorporated into the April 1982, revision. Accordingly, the staff concludes that it is appropriate to focus its review on issues of operating experience identified by the NRC since January 1980. As stated above, the

applicant reviewed and reported on the bulletins and GLs issued by the NRC between January 1, 1980, and December 31, 2002, as to their applicability to the AP1000 design.

As discussed in Section 20.7.1 above, the bulletins and GLs address the issues that are of sufficient safety significance to warrant requiring licensees to inform the NRC of the actions they have taken or will take, whereas information notices do not require a response. Accordingly, the NRC staff reviewed the bulletins and GLs issued between January 1, 1980, and December 31, 2002, applicable to the AP1000 design.

Upon initial review, certain bulletins and GLs were excluded from the review because they were not relevant to the design of the AP1000 plant, or because they were associated with TMI Action Plan items, USIs or GSIs, or existing rules and regulations and, thus, were already an integral part of the staff's AP1000 design review process.

The resolution of the technically relevant generic issues in NUREG-0933 (i.e., TMI Action Plan items, USIs, and GSIs) for the AP1000 design are addressed in Sections 20.2 through 20.4 of this report. The resolution of the issues in bulletins and GLs is summarized in Tables 20.7-1 and 20.7-2, respectively, of this report.

20.7.4 Conclusion

The bulletins and GLs issued by the NRC between January 1, 1980, and December 31, 2002, and incorporated into the staff review of the AP1000 design because the issues involved in these documents were not already required by rule, regulation, or policy statement. These are listed in Tables 20.7-1 and 20.7-2 of this report.

On the basis of its review of the bulletins and GLs issued between January 1, 1980, and December 31, 2002, and the applicant's report (WCAP-15800) on how these bulletins and GLs apply to the AP1000 design, the staff concludes that the applicant has adequately addressed the incorporation of operational data into the AP1000 design, except as noted in this report.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1, 1980, and December 31, 2002, for the Westinghouse AP1000 Design

Bulletin No. and Title	Staff Resolution
BL-80-01, Operability of ADS Valve Pneumatic Supply	This bulletin was issued only to BWR licensees to determine the operability of the pneumatic operator for the ADS; however, the AP1000 design has an ADS similar to BWRs. In WCAP-15800 (Rev. 1), the applicant states that 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 non-safety-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 the subject of pre-operational and in-service testing, and they will be included in the reliability assurance program. The staff agrees with the above information. Therefore, this bulletin is not applicable to the AP1000 design

Bulletin No. and Title	Staff Resolution
BL-80-04, Analysis of a Pressurized-Water Reactor (PWR) Main Steamline Break with Continued Feedwater Addition	The staff considered this bulletin in its review of DCD Tier 2 Section 15.1.5, and DCD Tier 2 Section 6.2.1.4 on mass and energy release analysis for a postulated pipe rupture inside containment. This bulletin asked addressees to review their containment pressure and temperature response analysis to determine if the main steamline break accident inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources such as continuation of feedwater or condensate flow. It also asked addressees to consider the ability to detect and isolate the damaged SG from these sources. In DCD Tier 2 Sections 6.2.1.4.1.3 and 6.2.1.4.3.2, the applicant indicates that the effects of startup feedwater flow are maximized in the main steamline break mass and energy release by assuming maximum (runout) startup feedwater flow to a fully depressurized SG starting from the safeguard system signal or low SG level reactor trip and continuing until automatically terminated. Regarding normal feedwater, the applicant indicated in DCD Tier 2 Section 6.2.1.4.1.2, that the unisolated feedwater line volumes between the SG and isolation valves has been accounted for in the mass and energy release. The feedwater flow rates are based on steam and main feed system design. Feedwater is isolated on a containment pressure signal. Because normal and startup feedwater addition have been maximized and because the AP1000 has means to automatically isolate feedwater flow, the staff finds that the licensee has adequately addressed the containment-related issues in Bulletin 80-04. The containment-related aspects of Bulletin 80-04 are therefore resolved. The other aspect of the feedwater addition issue addressed by this bulletin, namely the reactivity addition that would occur as a result of a main steamline break, is addressed in Section 15.2.1.5 of this report. The reactivity-related aspects of this bulletin are considered resolved based on the staff’s acceptance of the analyses provided in DCD Tier 2 Section 15.1.5. Based on the staff review, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL-80-05:Vacuum Condition Resulting in Damage To Chemical Volume Control System (CVCS) Holdup Tanks	Bulletin 80-05, "Vacuum Conditions Resulting in Damage to CVS Holdup Tanks (sometimes called Clean Waste Receiver Tanks)," addresses the issues concerning the release of radioactive material or other adverse effects as a result of low vacuum conditions causing tank buckling. The low-vacuum condition is created by the cooling of hot water in a low-pressure tank. NUREG-1512, adequately addressed the concern identified in IEB 80-05. The bases for the finding are (1) except for the reactor coolant drain tank (RCDT) located in the containment building, no other tank in the WLS is exposed to hot water, and (2) the RCDT has several design features, including an external design pressure of 15 psig, which eliminate the possibility of structural collapse of the RCDT resulting from steam condensation. Because of these design features, the RCDT will not collapse even if it is exposed to a full vacuum. The staff noted that all of the WLS tanks have vents that are adequately sized to prevent tank collapse during drain down. DCD Tier 2 Table 11.2.2 shows the external design pressure for the RCDT of the AP1000 design is 15 psig. The staff confirmed that the above bases for the finding in NUREG-1512 are applicable to AP1000, in Section 11.2 of this report. Therefore, the staff finds that the design of the WLS of AP1000 as discussed adequately addresses the concern identified in IEB 80-05 and is, therefore, acceptable. Based on the staff review, this bulletin is resolved for the AP1000
BL 80-06, Engineered Safety Feature Reset Controls	Westinghouse stated in WCAP-15800 that this bulletin is addressed in DCD Tier 2 Sections 7.3.1.1, and 13.5, and Chapter 14. This bulletin listed the following two actions that apply to the AP1000 design: (1) review the I&C system schematics to verify the ESF equipment remains in its emergency mode upon reset of the ESF actuation signal and (2) verify the as-built I&C system configuration conforms with schematics. For the AP1000 design, resetting the ESF signal does not reposition any ESF equipment. Verification of the as-built I&C system is the responsibility of the COL applicant during the plant pre-operational tests. This is COL Action Item 20.7-1. The last action required by the bulletin is plant-specific and does not apply to the AP1000 design. This bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 80-08, Examination of Containment Liner Penetration Welds	Westinghouse stated in WCAP-15800, Rev. 1 that the bulletin is not applicable to the AP1000 design because the design has no containment liner. The staff agrees with this assessment that this bulletin does not apply to the AP1000 design.
BL-80-10, Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to Environment	The event described in this bulletin was caused by the use of a temporary heating hose, which resulted in contamination of a nonradioactive system and an unmonitored, uncontrolled release of radioactivity to the environment. In DCD Tier 2 Section 9.3.5, the applicant states that there are no permanent connections between the WRS and non-radioactive piping. However, provisions are included for temporary diversion of contaminated water from normally nonradioactive drains to the WLS. Therefore, the WRS is designed to prevent the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal. WCAP-15800 (Rev.1) states that this bulletin was not applicable to the AP1000 and is the responsibility of the COL applicant. The staff does not believe that such an event is caused by poor system design but because of poor system operation and maintenance programs. Therefore, the staff agrees with the applicant that the COL applicant should address this event in its plant operating and maintenance procedures. Therefore, this bulletin is resolved for the AP1000 design.
BL 80-11, Masonry Wall Design	As stated in DCD Tier 2 Section 3.8.4.6.1.4, there are no safety-related masonry walls used in the nuclear island. Also, in WCAP-15800, Rev. 1, Westinghouse stated that this bulletin is not applicable to the AP1000 design because the design has no safety-related masonry walls. The staff agrees that the AP1000 has no safety-related masonry walls, and concludes that this bulletin is not applicable to the AP1000.

Bulletin No. and Title	Staff Resolution
BL-80-12, Decay Heat Removal Operability	This bulletin dealt with reducing the likelihood of losing the decay heat removal capability in operating PWRs. In WCAP-15800, Rev. 1, “Operational Assessment for AP1000,” dated December 2002, the applicant stated that this bulletin is addressed in DCD Section 7.4.1. The AP1000 relies on the passive RHR system for decay heat removal. For defense-in-depth considerations, AP1000 relies on RNS and associated procedures to reduce the shutdown mode risks. The staff evaluations of the PRHR capability and the shutdown risks involving RNS, respectively, are discussed in Sections 6.3 and 19.3 of this report. This bulletin is resolved for AP1000.

Bulletin No. and Title	Staff Resolution
BL-80-15, Possible Loss of Emergency Notification System (ENS) With Loss of Offsite Power	BL-80-15 directs licensees to take various Emergency Notification System (ENS) inspection and testing actions, including preparing an administrative procedure, and making necessary modifications to ensure that the ENS is not lost upon loss of either onsite or offsite power. In DCD Tier 2 Section 8.2.5 ("Combined License Information for Offsite Electrical Power"), the applicant states that "Combined License applicants referencing the AP1000 certified design will address the design of the ac power transmission system and its testing and inspection." DCD Tier 2 Section 13.3 ("Emergency Planning") states that emergency planning, including communication interfaces among the main control room, the technical support center and the emergency planning centers, are the responsibility of the COL applicant. Further, the COL applicant referencing the AP1000 certified design will address emergency planning, including post 72-hour actions and its communications interface. DCD Tier 2 Section 9.5.2 ("Communication System") provides that COL applicants referencing the AP1000 certified design will address interfaces to required offsite locations. Further, the emergency response facility communication system, including the crisis management radio system, will be addressed by the COL applicant (subsection 9.5.2.5). Since the ENS is an offsite emergency communications interface with the NRC, and communication system and interfaces (including the design, inspection and testing of the electric power systems) are the responsibility of the COL applicant, the staff finds that BL-80-15 is not applicable to the AP1000 design. The reminder to the COL applicant to review this bulletin for recommendations related to loss of either onsite or offsite power, and a consequential loss of the ENS is COL Action Item 20.7-2.

Bulletin No. and Title	Staff Resolution
BL-80-18, Maintenance of Adequate Minimum Flow Through Centrifugal Charging Pumps Following Secondary-Side, High-Energy-Line Rupture	BL-80-18 recommended modification to equipment and/or procedures, if calculations determine necessary, to assure adequate minimum flow through the centrifugal charging pumps under all conditions. In WCAP-15800, Rev. 1, Westinghouse stated that this bulletin was not applicable to the AP1000 design because the AP1000 design has no safety-related charging pumps as part of safety injection. The staff agrees that this bulletin is not applicable to the AP1000 design. Therefore, this bulletin is resolved.
BL 80-20, Failure of the applicant Type W-2 Spring Return to Neutral Control Switches	The applicant stated that this bulletin was not applicable to the AP1000 design because it involved a procurement issue. The staff agrees that the issues in this bulletin involve procurement and are the responsibility of the COL applicant. This is as part of COL Action Item 20.7-3. Based on the staff review, this bulletin is resolved for the AP1000
BL-80-24, Prevention of Damage Due to Water Leakage Inside Containment	Bulletin 80-24 defines an open system as one that utilizes an indefinite volume, such as a river, so that leakage from the system could not be detected by inventory decrease. The applicant stated that there are no open systems in the AP1000 containment. Cooling water for the AP1000 design is supplied by closed systems, including the component cooling water system (DCD Tier 2 Section 9.2.2) and the chilled water system (DCD Tier 2 Section 9.2.7). Fire protection water used inside containment is stored in the passive containment cooling water storage tank (PCCWST), and isolated by containment isolation valves during operation. Water level in the PCCWST is alarmed in the MCR and excessive flow from the tank can be terminated. Monitoring containment sump level is a key part of AP1000 leakage detection, which provides assurance that an increasing sump water level will be detected. DCD Tier 2 Section 5.2.5 provides a description of leakage detection and DCD Tier 2 Section 3.4.1.2.2.1 addresses containment flooding events. Based on the staff review, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 81-01, Revision 1 Surveillance of Mechanical Snubbers	The staff review of the resolution comment for this item in WCAP-15800, Revision 0, found that the reference to DCD Tier 2 Section 3.9.6 did not provide the appropriate discussion for resolution of this issue. DCD Tier 2 Section 3.9.6 addresses only inservice testing of pumps and valves, and does not include any information on mechanical snubbers. The staff recognizes that I.E. Bulletin 81-01 dealt with examinations of snubbers installed in operating plants, and this aspect of the bulletin is not applicable to the AP1000 design certification. However, the staff position is that the surveillance testing implications of this bulletin should be addressed during the design certification process. DCD Tier 2 Section 3.9.6 does not provide this information. The staff requested, in RAI 210.068, that the applicant provide additional discussion which address surveillance and testing of dynamic restraints, i.e. snubbers, used in the AP1000 design. In response to RAI 210.068, the applicant referenced DCD Tier 2 Section 3.9.3.4.3 for the discussion of requirements for the production and qualification of hydraulic snubbers. Additionally, DCD Tier 2 Section 5.2.4 states that inservice inspection and testing of Class 1 components, including snubbers used as supports, are performed in accordance with Section XI of the ASME Code. ASME Code, Section XI references the ANSI/ASME OM Code, Part 4 standard for inservice examinations and inservice testing of snubbers. DCD Tier 2 Section 3.9.8.3 identifies the requirement for the COL applicant to develop a program to verify the operability of snubbers utilized in the AP1000 design. In Revision 3 of DCD Tier 2 Section 3.9.3.4.3, the applicant added specific references to the ASME OM Code used to develop the inservice testing plan for the AP1000 Design Certification, and Section XI of the ASME Code for performance of inservice testing. WCAP-15800, Revision 1 provided appropriate references to DCD Tier 2 Section 3.9.3.4.3 and to ASME Code, Section XI for information addressing snubber surveillance testing. The staff review of this information concludes that the changes in Revision 1 of the WCAP adequately address this issue, and provide an acceptable resolution for this IE Bulletin by ensuring that programs are established for qualification testing of snubbers, and inservice examination and functional testing of snubbers. Based on the staff review, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 81-02, Failure of Gate Type Valves to Close Against Differential Pressure	In WCAP-15800, Rev. 1, the applicant references DCD Tier 2 Sections 3.9.6.2, 5.4.8.1.2, and 5.4.8.2 as the basis for the resolution of this bulletin. The staff agrees with this basis. Because the subject of this bulletin led, in part, to the issuance of GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," the staff's position is that the basis for disposition of BL-81-02 should be addressed in DCD Tier 2 . As discussed in Section 3.9.6.2 of this report, the staff has concluded that the commitments in DCD Tier 2 Section 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis to resolve this issue. Based on the staff review, this bulletin is resolved for the AP1000 design.
BL-81-03, Flow Blockage of Cooling Water to Safety System Components by Corbicula SP (Asiatic Clams) and Mytilus SP (Mussels)	The applicant stated that this bulletin was not applicable to the AP1000 design because the AP1000 does not depend on a site water intake structure for safety-related heat removal. The staff agrees with the applicant that this bulletin is not applicable to the AP1000 because the component coolant system and service water system are not used for cooling safety-related components. In addition, service water strainers and service water chemical injection are addressed in DCD Tier 2 Section 9.2.1.2.2. Therefore, this bulletin is not applicable to AP1000.
BL 82-02, Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants	Westinghouse stated in WCAP-15800, Rev. 1, that this bulletin was not applicable to the AP1000 design. Also, the use of lubricants containing molybdenum disulfide are specifically prohibited for use in the AP1000 design by DCD Tier 2 Section 5.2.3.5. Based on the staff review, this bulletin is resolved for the AP1000 design.
BL-83-03, Check Valve Failures in Raw Water Cooling Systems of Diesel Generators	In WCAP-15800, Rev. 1, Westinghouse stated that this issue is not applicable because the AP1000 diesel generators have no safety-related functions. The staff agrees with the applicant position that BL-83-03 does not apply to the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL-84-03, Refueling Cavity Water Seal	This bulletin is not applicable to the AP1000 design because the design does not use this type of seal as discussed in, DCD Tier 2 Section 1.2.1.2.1. The AP1000 uses a permanent welding seal ring between the vessel flange and the refueling cavity floor. This bulletin is resolved for the AP1000 design.
BL 85-02, Undervoltage Trip Attachments of the applicant DB-50 Reactor Trip Breaker	Westinghouse stated in WCAP-15800 that this bulletin is addressed in DCD Tier 2 Sections 7.1.2.2.4 and Chapter 16, Section 3.3.1.6. This bulletin (1) assured proper reactor trip breaker (RTB) testing in plants that had not yet installed the automatic shunt trip modification and (2) provided information about RTB reliability and TS operability. The AP1000 design addresses this first part by providing automatic diverse trip actuation via the shunt trip attachment. Testing of the interface allows trip actuation of the breakers by either the undervoltage trip attachment or the shunt trip attachment. The applicant also provided sufficient information on RTB reliability and TS operability to adequately address the second part of the bulletin. Based on the staff review, this bulletin is resolved for the AP1000.
BL 85-03, Motor-Operated Valve Common Mode Failures during Plant Transients due to Improper Switch Settings	In WCAP-15800, Rev. 1, the applicant references DCD Tier 2 Section 3.9.6.2 as the basis for the resolution of this bulletin. The staff agrees with this basis. Because the subject of this bulletin led, in part, to the issuance of GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," the staff's position is that the basis for disposition of BL-85-03 should be addressed in the DCD Tier 2. As discussed in Section 3.9.6.2 of this report, the staff has concluded that the commitment in DCD Tier 2 Section 3.9.6 relative to inservice and qualification testing of MOV provides an acceptable basis to resolve this issue.
BL-86-01, Minimum Flow Logic Problems That Could Disable Residual Heat Removal (RHR) Pumps	This bulletin recommended BWR licensees or applicants to provide appropriate instructions and training to operators to deal with the loss of RHR pumps caused by a single-failure of the isolation valve in the mini-flow lines for the pumps. In WCAP-15800, Rev. 1, Westinghouse stated that this bulletin was not applicable to the AP1000 design because AP1000 has no valves in the mini-flow lines for the normal RHR system (RNS), and the RNS pumps have no safety-related function. The staff agrees with the applicant, and this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL-86-03, Potential Failure of Multiple ECCS Pumps Due to Single-Failure of Air-operated Valve in Minimum-flow Recirculation Line	Westinghouse stated, in WCAP-15800 (Rev. 1), that this bulletin is not applicable to the AP1000 design because the design does not have valves in mini-flow lines. The staff has reviewed this issue and agrees with the applicant that this bulletin is not applicable to the AP1000 design. Based on the staff review, this bulletin is resolved for the AP1000
Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants	In WCAP-15800, Revision 1, “Operational Assessment for AP1000,” the applicant indicates that this bulletin is not applicable to the AP1000 design because it is a surveillance issue as discussed in DCD Tier 2 Sections 5.4.3.4 and 10.3.6. The purpose of this bulletin was to request that licensees submit information concerning their programs for monitoring the thickness of pipe walls in high-energy single-phase and two-phase carbon steel piping systems. Licensees were requested to provide specific information concerning their programs for monitoring the wall thickness of pipes in condensate, feedwater, steam, and connected high-energy piping systems, including all safety-related and non-safety-related piping systems fabricated of carbon steel. DCD Tier 2 Section 5.4.3.4 pertains to RCS piping which is fabricated from stainless steel and, therefore, is not addressed by Bulletin 87-01. DCD Tier 2 Section 10.1.2 discusses steam and power conversion system piping design and pipe wall thickness inspections for erosion/corrosion protection. DCD Tier 2 Section 10.1.3 indicates the COL holder will address preparation of an erosion/corrosion monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam. Based on this information, the staff concludes that this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 87-02, Fastener Testing to Determine Conformance with Applicable Material Specifications	The purpose of this bulletin is to request that licensees 1) review their receipt inspection requirements and internal controls for fasteners and 2) independently determine, through testing, whether fasteners (studs, bolts, cap screws and nuts) in stores at their facilities meet required mechanical and chemical specification requirements. In DCD Tier 2 Section 1.9.5.5, “Operational Experience,” the applicant states: 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, “Operational Assessment for the AP1000." In WCAP-15800, “Operational Assessment for AP1000,” page 3-17 Table 2; “I.E. Bulletin,” page 2-5; and NRC Bulletin 87-02, Fastener Testing to Determine Conformance with Applicable Material Specifications, the applicant states this issue is related to procurement and is not applicable to the AP1000 design. The NRC staff agrees with the applicant that Bulletin 87-02 is not applicable to the AP1000 design certification review since this is a procurement issue. The COL applicant is responsible for procurement issues.
BL 88-01, Defects of the applicant Circuit Breakers	In DCD Tier 2 the applicant states this issue is related to procurement and is not applicable to the AP1000 design. The NRC staff agrees with the applicant that Bulletin 87-02 is not applicable to the AP1000 design certification review since this is a procurement issue. The COL applicant is responsible for procurement issues.
BL-88-04, Potential Safety-Related Pump Loss	Westinghouse stated, in WCAP-15800 (Rev. 1), that the bulletin was not applicable to the AP1000 design because the design has no safety-related pumps. The safety-related cooling systems are passive systems. The staff reviewed this issue and agrees with this conclusion. Therefore, this bulletin is not applicable to the AP1000 design

Bulletin No. and Title	Staff Resolution
BL 88-08 Thermal Stresses in Piping Connected to RCSs	WCAP-15800, Rev. 1, stated that this bulletin is addressed in DCD Tier 2 Sections 3.9.3.1.2 . The staff concluded that the information in DCD Tier 2 Section 3.9.3.1.2 provides an acceptable basis for resolving BL-88-08 for the AP1000. The staff’s evaluation of this issue is in Section 3.12.5.9 of this report. Based on the staff review, this bulletin is resolved for the AP1000 design.
BL 88-09 Thimble Tube Thinning in the applicant Reactors	The staff reviewed description of the design of the AP1000 thimble tubes. Given in DCD Tier 2 Section 3.9.7.2, that has enhanced resistance to flow-induced vibration and wear. The thimble tube is structurally stiffer than the design in previous operating plants, utilizes wear resistant materials, and features a smaller gap between the thimble tube and the thimble guide tube to further minimize vibration. The double-wall design of the thimble tube assembly also precludes a non-isolable leak of reactor coolant. The staff review concludes that the enhanced design of the AP1000 incore instrumentation thimble tubes adequately addresses the BL 88-09 concerns for accelerated wear of incore thimble tubes in the applicant operating reactor designs. Based on the staff review, this bulletin is resolved for the AP1000 design.
BL 88-11, Pressurizer Surge Line Thermal Stratification	WCAP-15800, Rev. 1, stated that this bulletin is addressed in DCD Tier 2 Sections 3.9.3.1.2. The staff concluded that the information in DCD Tier 2 Section 3.9.3.1.2 provides an acceptable basis for resolving BL 88-11 for the AP1000. The staff’s evaluation of this issue is provided in Section 3.12.5.10 of this report. Based on the staff review, this bulletin is resolved for the AP1000 design.
Bulletin 89-01, Failure of the applicant SG Tube Mechanical Plugs	In WCAP-15800, Revision 1, “Operational Assessment for AP1000,” the applicant indicated that this bulletin is not applicable to the AP1000 design because the issue involves procurement. The staff agrees with this assessment since plugs installed into the SG following operation will be purchased by the COL applicant. Based on the staff review, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL-89-03, Potential Loss of Required Shutdown Margin During Refueling Operations	BL-89-03 requires licensees to take actions to prevent potential loss of required shutdown margin during the movement and placement of highly reactive fuel during refueling operation. In WCAP-15800, Rev. 1, Westinghouse stated that this bulletin is not applicable to the AP1000 design because this is a procedural issue. The applicant also refers this issue to DCD Tier 2 Section 13.5.1, "Combined License Information Item," and 4.3.1.5, "Shutdown Margins." The staff agrees that this bulletin involved procedures; however, these procedures would involve movement and placement of highly reactive fuel during refueling within the core designed by the applicant. DCD Tier 2 Section 9.1 discusses fuel storage and handling, including the refueling equipment used to safely move and store fuels. Additionally, the IRWST provides large quantities of borated water that maintains the required shutdown margin. DCD Tier 2 Subsection 9.1.6 also describes the responsibility of the COL applicant, which is designated as COL Action Item 20.7-4. Based on the staff review, this bulletin is resolved for the AP1000.
BL 90-01, Loss of Fill-Oil in Transmitters Manufactured by Rosemount	The applicant states in WCAP-15800 that this bulletin is not applicable to the AP1000 design because it involves a procurement issue. Supplement 1 to this bulletin states that transmitters manufactured after July 11, 1989, are not subject to the fill-oil leakage problems identified in the bulletin. The staff agrees that this bulletin is resolved for the AP1000 design.
BL-92-01, Failure of Thermo-lag 330 Fire-barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage	As stated in DCD Tier 2, Thermo-Lag is not used in the AP1000 design, therefore, this bulletin is not applicable to the AP1000 design. Based on the staff review, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL-93-02, Debris Plugging of Emergency Core Cooling Suction Strainers	This bulletin deals with the installation or storage of fibrous air filters or other temporary sources of fibrous material in containment that are not designed to withstand a LOCA. The applicant stated in a letter dated April 9, 2003, that the AP1000 has no ventilation filters inside containment. This satisfies the intent of BL 93-02 and resolves it for the AP1000 design. Based on the staff review, this bulletin is resolved for the AP1000 design.
BL-95-02, Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode	This bulletin deals with the need for BWR licensees to ensure that their suppression pools are relatively free of debris that could clog the suction strainers of safety-related pumps which take suction from the suppression pool. In addition, the bulletin requests that BWR licensees determine whether there are adequate controls to ensure that foreign material exclusion (FME) procedures are effective. The AP1000's IRWST serves several functions similar to those of BWR suppression pools. For example, it provides a source of cooling water to the reactor core along with the core makeup tanks and the accumulators. In addition, the first three stages of the ADS discharge to the IRWST. The IRWST is made of stainless steel and thus, would not constitute a significant source of corrosion products. Piping lines leading to the IRWST are also made of stainless steel or are stainless steel clad. Normally, closed louvers are designed to prevent any debris from entering the IRWST through overflow and vent lines during normal operation. DCD Tier 2 Section 6.3.8.1 states that " COL Applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages." This satisfies the FME aspects of this bulletin. Based on the staff review, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL-96-01, Control Rod Insertion Problems	This bulletin required PWR licensees to assess the operability of control rods because of the problems of incomplete control rod insertion (IRI) encountered in some PWR plants. In WCAP-15800, Rev. 1, Westinghouse stated that this bulletin is not applicable to the AP1000 design because it is a procedural issue. It has been determined that the IRI was caused by thimble tube distortion resulting from excessive load. Because this is a fuel design problem, and the applicant has not committed to any fuel manufacturers, the staff concluded that the applicant does not have to address this issue, unless it has committed to certain fuel designs discussed in the bulletin. This issue should be appropriately addressed by the COL applicant. This is COL Action Item 20.7-5, Based on the staff review, this bulletin is resolved for the AP1000
BL-96-02, Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety-Related Equipment	This bulletin reminded licensees of their responsibilities for ensuring that activities involving the movement of heavy loads are performed safely. It also requested that licensees review their plans and capabilities for handling heavy loads and assure that their load handling operations are in accordance with existing regulatory guidelines and the licensing basis. This issue is addressed by the resolution of USI A-36 in DCD Tier 2 Section 1.9.4., which states that the AP1000 design conforms to NUREG-0612 and Section 9.1.5 of the SRP. The staff determined that ensuring the safe movement of heavy loads is the responsibility of the COL applicant. Based on the staff review, this bulletin is resolved for the AP1000

Bulletin No. and Title	Staff Resolution
BL-96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors	This bulletin provides the final resolution of the ECCS suction strainer blockage issue for operating BWRs. The resolution includes the option of installing large passive suction strainers. The staff considers that the applicant has addressed this bulletin in large measure through controlling potential sources of debris (e.g., prohibiting the presence of installed fibrous material in zones of containment vulnerable to jet impingement and flooding, constructing the IRWST of stainless steel, and requiring a containment cleanliness program). The staff concludes that the applicant has adequately addressed the root causes of strainer blockage that were identified in this bulletin and considers it to be resolved for the AP1000 design. However, the staff is currently in the process of resolving a similar suction screen blockage issue for the current generation of PWRs in conjunction with GSI 191. Section 6.2.1.8 of this report provides staff's evaluation of the design of the IRWST and containment recirculation screens in the context of Issue 191.
Bulletin 2001-01, Circumferential Cracking of RPV Head Penetration Nozzles	In WCAP-15800, Revision 1, "Operational Assessment for AP1000," Westinghouse indicated that the resolution of this bulletin is contained in DCD Tier 2 Section 5.2.3, "Reactor Coolant Pressure Boundary Materials." This section of the DCD Tier 2 indicates that the use of nickel-chromium-iron alloy in the AP1000 reactor coolant pressure boundary in contact with the reactor coolant is limited to Alloy 690/52/152 materials. While the staff agrees that the use of Alloy 690/52/152 materials is an improvement, the staff does not consider that this satisfactorily addresses all aspects of Bulletin 2001-01. As part of its review of DCD Tier 2 Section 4.5.1, "Control Rod Drive System Structural Materials," the staff reviewed the pertinent aspects of this bulletin as they apply to the design, fabrication, and inspection of control rod drive nozzle penetrations. The staff's evaluation of this information is contained in Section 4.5.1 of this report. Pending the resolution of open items in Section 4.5.1 of this report, this bulletin is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	This bulletin was issued to obtain information needed to determine the adequacy of pressurized water reactor (PWR) plants' boric acid corrosion control (BACC) programs. Within 60 days of the date of this bulletin, all PWR addressees were required to submit to the NRC the basis for concluding that their boric acid corrosion control program for the reactor coolant pressure boundary (RCPB) is providing reasonable assurance of compliance with the applicable regulatory requirements. Bulletin 2002-01 indicated that the information submitted will be used by the staff to determine the need for, and to guide the development of, additional regulatory actions to address degradation of the reactor pressure vessel head and/or other portions of the reactor coolant pressure boundary. Experience with currently operating PWRs continues to show cracking of Alloy 600 components. Recent experience appears to indicate that cracking has even occurred in welds or components not previously expected to crack based on the temperature of the weld or component and the time in service. The staff believes that the use of Alloy 690 materials in contact with reactor coolant is a substantial improvement over the use of materials currently in wide use in the industry. However, data is not presently available to demonstrate that cracking in these welds and components will not occur over the projected 60 year lifetime of an AP1000 plant. Bare metal visual inspection of these locations is highly effective in identifying locations where cracking occurs. Technical specification requirements prohibit through wall leakage of the RCPB. Therefore, Westinghouse needs to provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions. This is Open Item 20.7-1. In addition, as noted in the bulletin, the staff is considering the need for additional regulatory actions to ensure that an effective program is in place to monitor potential cracking of these susceptible materials and ensure that the causes of cracking are appropriately addressed. If the staff develops new monitoring requirements, the staff will consider the need to backfit these requirements on operating reactors and certified designs, including the AP1000.

Bulletin No. and Title	Staff Resolution
BL 2002-02, Reactor Pressure Vessel Head Penetration Nozzle Inspection Programs	This bulletin was issued after WCAP-15800, Revision 1, “Operational Assessment for AP1000,” was issued. The staff’s review of this application for Bulletin 2002-01 relied on information contained in DCD Tier 2 Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.” This section of the DCD Tier 2 indicates that the use of nickel-chromium-iron alloy in the AP1000 reactor coolant pressure boundary in contact with the reactor coolant is limited to Alloy 690/52/152 materials. While the staff agrees that the use of Alloy 690/52/152 materials is an improvement, the staff does not consider that this satisfactorily addresses all aspects of Bulletin 2002-01, and that additional inspections will be needed. As part of its review of DCD Tier 2 Section 4.5.1, “Control Rod Drive System Structural Materials,” the staff reviewed the pertinent aspects of this bulletin as they apply to the design, fabrication, and inspection of control rod drive nozzle penetrations. The staff’s evaluation of this information is contained in Section 4.5.1 of this report. Pending the resolution of open items in Section 4.5.1 of this report, this bulletin is resolved for the AP1000 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 2002, for the Westinghouse AP1000 Design

GL No. and Title	Staff Resolution
GL-80-01, Report on ECCS Cladding Models	GL 80-001 informed all licensees about an extension of one week for written comments to the draft NUREG-0630, “Cladding, Swelling and Rupture Models for LOCA Analysis.” This administrative communication is not applicable to the AP1000 design. However, NUREG-0630 is applicable to AP1000. As indicated in WCAP-15800, Rev. 1, “Operational Assessment for AP1000,’ dated December 2002, the NUREG-0630 cladding, swelling and rupture models are included in WCAP-12945, “ Westinghouse Code Qualification Document for Best Estimate Loss-of-Coolant Accident Analysis.” The LOCA analysis is addressed in DCD Tier 2 Chapter 15. Based on the staff review, this issue is resolved for the AP1000 design
GL-80-02, Quality Assurance Requirements Regarding Diesel Generator Fuel Oil	This GL was concerned with requirements on DG fuel oil in the QA program. In WCAP-15800 (Rev. 1), Westinghouse stated that this GL is not applicable to the AP1000 design, because the AP1000 does not have safety-related diesel generators, as discussed in DCD Tier 2 Section 8.3.1. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-09, Low-Level Radioactive Waste Disposal	This GL concerned the requirements for solid waste shipments from a plant. To the extent that GL-80-009 applies to the design of AP1000, it is addressed in DCD Tier 2 Section 11.4.2. In addition, to ensure the COL applicant conforms to GL-80-009, DCD Tier 2 Section 11.4.6, "Combined License Information for Solid Waste Management System Process Control Program," identifies the GL as a part of COL Action Item 11.4-1. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-80-013, Qualification of Safety-Related Equipment	This GL concerned the adequacy of the electrical equipment environmental qualification program. This issue is not considered relevant to the design review for the AP1000 because it deals with auditing the documents of the qualification program. Based on the staff review, this GL is resolved for the AP1000 design.
GL 80-014, LWR Primary Coolant System Pressure Isolation Valves	In WCAP-15800, Revision 1, Westinghouse stated that DCD Tier 2 Section 1.9.4.2.2, Issue USI-B-63, discusses this issue. The staff review of DCD Tier 2 Section 1.9.4.2.2 concludes that the AP1000 plant incorporates appropriate isolation and adequate design of low-pressure systems that interface with high-pressure systems. The staff's evaluation of this issue is provided in Section 3.9.3.1 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-016, IEB 79-01b Environmental Qualification of Class 1E Equipment	This GL concerned meetings held on Bulletin 79-01 and questions about environmental qualification. This issue is not relevant to the design review for the AP1000. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-019, Resolution of Enhanced Fission Gas Release Concern	In WCAP-15800, Rev. 1, Westinghouse stated that no action was required for AP1000 by this GL. However, the fission gas release models for the AP1000 design are accounted for in the fuel performance code discussed in WCAP-10851-P-A and WCAP-11873-A, "Improved Fuel Performance Models for the applicant Fuel Rod design and Safety Evaluations." Based on the staff review, this issue is resolved for the AP1000 design.
GL-80-026, Qualification of Reactor Operators	This generic letter set forth revised criteria to be used by the staff in evaluating reactor operator training. Westinghouse stated that this generic letter is the responsibility of the COL applicant. Based on the staff review, this issue is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-80-030, Clarification of the Term "Operable" as it Applies to Single-failure Criterion for Safety Systems Required by TS	Westinghouse stated that the definition of operable is addressed in Section 16.1 of DCD Tier 2 on plant TSs. By adoption of the improved Westinghouse Standard TS (STS), Westinghouse should adequately address the TS issues in this GL for the AP1000 design. The TSs for the AP1000 design are in DCD Tier 2 Chapter 16 and discussed in Chapter 16 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-035, Effect of a dc Power Supply Failure on ECCS Performances	The applicant stated that this GL is not applicable to the AP1000 design because it concerned only BWR plants; however, the effect of dc power supply failure on ECCS performance could apply to PWR. This GL addresses the concerns that the loss of a dc power supply could disable several ECCS components and, thereby, could result in a limiting single-failure conditions for some breaks. In WCAP-15800, Revision 1, Westinghouse stated that this GL is addressed in DCD Tier 2 Section 8.3.2, Table 8.3.2.7, "Failure Modes and Effects Analysis." The staff evaluated Section 8.3.2 and concluded that the effect of a dc power supply on ECCS is adequately addressed in Table 8.3.2.7. Therefore, GL 80-35 is resolved for the AP1000 design.
GL-80-045, Fire Protection Rule	This GL requested comments on a proposed rule adding a new Section 50.48 and Appendix R, which was attached to the GL. In a letter dated September 21, 1995, the staff informed the applicant that this issue was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-048, Revision to May 19, 1980 Letter on Fire Protection (GL-80-045)	See the resolution of GL-80-045 above.

GL No. and Title	Staff Resolution
GL-80-053, Decay Heat Removal Capability	This GL requested TS amendments concerning decay heat removal capability. The applicant stated that this GL is addressed in DCD Tier 2 Section 16.3.5, on plant TSs. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review. Therefore, this GL is resolved for the AP1000 design.
GL-80-077, Refueling Water Level	In Revision 1 of WCAP-15800, Westinghouse stated that this GL is not applicable to the AP1000 design and was the responsibility of the COL applicant. This is discussed in DCD Tier 2 Sections 13.5.1 and 13.5-2. The staff agrees with the applicant that this issue is the responsibility of the COL applicant. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-082, IEB 79-01b, Supp. 2, Environmental Qualification of Class 1E Equipment	See the resolution of GL-80-016 above.
GL 80-088, Seismic Qualification of Auxiliary Feedwater Systems	WCAP-15800, Revision 1 states that this issue is not applicable to the non-seismic portion of the AP1000 startup feedwater system (inside the turbine building), and that the safety-related portion of this system in the containment and auxiliary building is seismically qualified, as discussed in DCD Tier 2 Section 10.4.9. The staff agrees with these safety classifications, and concludes that the safety-related portion of the startup feedwater system is Seismic Category I. Based on the staff review, this GL is resolved for the AP1000 design.
GL-80-098, IEB 80-24, Prevention of Damage Due to Water Leakage Inside Containment	See the resolution of BL-80-24.

GL No. and Title	Staff Resolution
GL-80-099,TS Revisions for Snubber Surveillance	This GL released Revision 1 to the Inservice Surveillance Requirements for snubbers in the Standard Technical Specifications (STS) in 1980. The applicant improved STS eliminated this requirement. It was replaced by ANSI/ASME OM Part 4, which is now referenced in ASME Section XI for inservice testing and inspection of snubbers. As discussed under Bulletin 81-001 in this table and in Section 3.9.3.3 of this report, the applicant committed to ASME Section XI and ANSI/ASME OM Part 4. Therefore, GL-80-099 is not applicable to the AP1000.
GL-80-100, Appendix R to 10 CFR Part 50 Regarding Fire Protection	This GL stated that the Commission published a new Appendix R, fire protection, to 10 CFR Part 50. In a letter dated September 21, 1995, the staff informed the applicant that this issue was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL 80-109, Guidelines for SEP and Soil Structure Interaction Reviews	The AP1000 is designed for hard rock sites only, as indicated in WCAP-15800, Revision 1. The details of the staff review of the soil structure interaction issue are included in Section 3.7.2.4 of this report. The staff review of this issue concludes that soil structure interaction is not applicable to the AP1000, and does not consider soil structure interaction.
GL 81-014, Seismic Qualification of Auxiliary Feedwater Systems	WCAP-15800, Revision 1, states that this issue is not applicable to the non-seismic portion of the AP1000 startup feedwater system (inside the turbine building), and that the safety-related portion of this system in the containment and auxiliary building is seismically qualified (Ref. DCD Tier 2 Section 10.4.9). The staff review agrees with these safety classifications, and concludes that the safety-related portion of the startup feedwater system is Seismic Category I. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-81-021, Natural Circulation Cooldown	This GL addressed procedures and training to prevent, recognize, and react to reactor vessel voiding during natural circulation cooldown. In WCAP-15800, Rev. 1, the applicant refers this GL to emergency response guidelines (ERGs). The staff reviewed the existing AP600 ERGs, which are applicable to AP1000, determined that guidelines are sufficiently given to the operator to cool down the plant using natural circulation means. Based on the staff review, this GL is resolved for the AP1000 design.
GL-81-38:Storage of Low-Level Radioactive Wastes at Power Reactor Sites	This GL provided guidelines for the storage of low-level radioactive wastes at plant sites. The applicant stated in WCAP-15800, Revision 1 that this GL was not applicable to the AP1000 because it is the responsibility of the COL applicant. This is a site-specific issue because it will depend upon the available offsite storage space for low-level radioactive waste from the plant. This will be identified by the COL applicant if it proposes an onsite low-level radioactive waste storage facility to the NRC. The NRC would then evaluate the proposed facility against the criteria in GL-81-38. DCD Tier 2 Section 11.4-1 identifies GL-81-38 as a part of COL Action Item 11.4-1.
GL-81-39, NRC Volume Reduction Policy	This GL provided the Commission policy statement on reduction of low-level radioactive wastes at plant sites. To the extent that GL-81-39 applies to the design certification of AP1000, it is addressed in DCD Tier 2 Section 11.4.2.1. To ensure that the COL applicant will conform with GL-81-39, DCD Tier 2 Section 11.4.6 identifies this GL as a part of COL Action Item 11.4-1. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-82-04, Use of INPO [Institute of Nuclear Power Operations] SEE-IN Program	This GL recommended the INPO Significant Event Evaluation and Information Network (SEE-IN) program to screen the large volume of raw data pertaining to operational experience throughout the industry. The applicant included a discussion of the review of operating experience in the discussion of the resolution of TMI Action Plan Item I.C.5 in DCD Tier 2 Chapters 1 and 18. The staff found this discussion acceptable. Based on the staff review, this GL is resolved for the AP1000 design.
GL-82-09, Environmental Qualification of Safety-Related Electrical Equipment	The GL requested comments on the then new rule on environmental qualification and the proposed RG 1.89 to implement the new rule. This issue is not relevant to the design review for the AP1000 because this has been implemented and reviewed. (See DCD Tier 2 Sections 1.9 and 3.11). Based on the staff review, this GL is resolved for the AP1000 design.
GL-82-23, Inconsistency Between Requirements of 10 CFR 73.40(d) and Standard Technical Specifications (STS) for Performing Audits of Safeguards Contingency Plans	By adoption of the Westinghouse STS, the applicant adequately addressed the TS issues in this GL. The TS for the AP1000 design are in DCD Tier 2 Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed the applicant that this issue was no longer considered relevant to the design review. On this basis, the GL is resolved for the AP1000 design.
GL-82-39, Problems with Submittals of 10 CFR 73.21 Safeguards Information for Licensing Reviews	Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication to the licensees. This generic letter is not a design issue because site security is within the scope of the COL applicant. This includes the reporting of safeguards information for licensing reviews. On this basis, the GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-83-07, The Nuclear Waste Policy Act of 1982	This generic letter provided a copy of the Nuclear Waste Policy Act of 1982 and explained the requirements. Westinghouse stated that this generic letter is not applicable to the AP1000 design because it was an administrative communication to licensees. The Nuclear Waste Policy Act of 1982 requires licensees to have a contract with the Department of Energy (DOE) before receiving a license and is within the scope of the COL applicant. This issue is not a design issue and, therefore, GL-83-087 is resolved for the AP1000 design.
GL-83-11, Licensee Qualifications for Performing Safety Analyses in Support of Licensing Actions	This GL provided a generic set of guidelines that the NRC will use to accept the licensee's qualification to perform its own safety analyses using approved computer codes or methods to support licensing actions. In WCAP-15800, Rev. 1, the applicant states that the AP1000 design is performed under a QA program which is reviewed by the NRC. Chapter 21 of this report presents the staff's evaluation of the applicant's testing program and computer code verification. Based on the staff review, this GL is resolved for the AP1000 design. However, the staff identifies COL Action Item 20.7-6 that if a COL applicant chooses to perform its own safety analysis in the future, it would follow the guideline specified in GL-83-11, Supplement 1.
GL-83-13, Clarification of Surveillance Requirements for HEPA Filters Charcoal Adsorber Units in STS on ESF Cleanup Systems	The applicant stated that this GL is not applicable because the AP1000 design has no safety-related ventilation systems. By adoption of the applicants STS, the applicant adequately addressed the TS issues in this GL. The TS for the AP1000 design are in DCD Tier 2 Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review. Based on the staff review, this GL was resolved for the AP1000.

GL No. and Title	Staff Resolution
GL-83-14, Definition of Key Maintenance Personnel.	This GL has been satisfactorily addressed by the applicant as an administrative procedure, and not applicable. Therefore, this GL is resolved for the AP1000 design the AP1000.
GL-83-15 Implementation of RG 1.150, “Ultrasonic Testing of Reactor Vessel Welds during Preservice and Inservice Testing”	The applicant has stated in DCD Tier 2 that the AP1000 design conforms to the recommendations of RG 1.150. Based on the staff review, this GL is resolved for the AP1000 design.
GL-83-21, Clarification of Access Control for Law Enforcement Visits	The staff has not completed review of this item. It is a part of the Open Item 13.6-1.
GL-83-22, Safety Evaluation of "Emergency Response Guidelines"	This GL stated that the the applicant ERG program was acceptable and provided improved guidance for development of plant emergency operating procedures. In WCAP-15800, Rev. 1, Westinghouse stated that this GL is not applicable to the AP1000 design because it is the responsibility of the COL applicant. The staff also identified COL Action Item 18.9-2 for the COL applicant to develop plant-specific EOPs using the ERGs. The staff reviewed the the applicant ERG program, and documented its evaluation in Section 18.9.3 of this report. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-83-26, Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests	This GL provided guidance about TS for surveillance of the fuel impurity levels for diesel generators. Westinghouse stated that this GL is not applicable to the AP1000 design as discussed in DCD Tier 2 Section 1.9, and Appendix 1A, about the inapplicability of RG 1.137, "Fuel-Oil Systems for Standby Diesel Generators." Westinghouse states that the onsite diesel generators and associated fuel-oil systems are non-safety-related. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-83-27, Surveillance Intervals in STS	Westinghouse stated that this STS is addressed in DCD Tier 2 Chapter 16 and TS Section 3.0. The staff found that adoption of the applicants STS, Westinghouse adequately addressed the TS issues in this GL. The TS for the AP1000 design are in DCD Tier 2 Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-83-28, Required Actions Based on Generic Implications of Salem ATWS Event	This generic letter addressed certain intermediate-term actions to be taken by licensees as a result of the Salem ATWS events on the basis of NUREG-1000. Applicant stated that this generic letter is addressed in the DCD Tier 2 Section 7.1.2.2.4. The staff agrees. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review. Therefore, this generic letter is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-83-30, Deletion of STS Surveillance Requirement 4.8.1.1.2.D.6 for Diesel Generator Testing	The applicant stated that this GL is not applicable to the AP1000 design as discussed in DCD Tier 2 Section 1.9, and Appendix 1A, about the inapplicability of RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants." the applicant states that the onsite diesel generators are non-safety-related. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.
GL-83-32, NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS.	The applicant stated that this GL is addressed in DCD Tier 2 Section 18.8.2.1.2 The staff has completed its review of DCD Tier 2 Section 18.8.2.1.2 (Revision 3) and finds the applicant position acceptable. Based on the staff review, this GL is resolved for the AP1000 design.
GL-83-33, NRC Positions on Certain Requirements of Appendix R to 10 CFR Part 50	This GL is addressed in DCD Tier 2 Section 9.5.1. The staff included, in its review of the AP1000 design, the positions of the GL, and Based on the staff review, this GL is resolved for the AP1000 design.
GL 83-41, Fast Cold Start of Diesel Generator	The applicant stated in DCD Tier 2 that diesel generator for the AP1000 design are not safety-related. Therefore, This GL is not applicable for AP1000 design. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.
GL 84-04, Safety Evaluation of the Applicant Topical Reports on Elimination of Postulated Pipe Breaks in PWR Primary Main Loops	WCAP-15800, Revision 1, references DCD Tier 2 Section 1.9.4.2.2, USI A-2 for the response to this issue. The staff review of DCD Tier 2 Section 1.9.4.2.2, Task Action Plan Item A-2, concludes that the discussion of the application of mechanistic pipe break (leak-before-break) criteria for elimination of the analysis of the dynamic effects of a postulated instantaneous rupture of the AP1000 primary loop piping provides the basis for an acceptable resolution of GL-84-04. The staff review of the the applicant application of leak-before-break criteria is included in Section 3.6.3 of this report. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-84-09, Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii)	This issue remains open because DCD Tier 2 for the control of combustible gas in containment during accidents does not comply with current regulations. The NRC has proposed major changes to 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,” and related changes to 10 CFR 50.34 and 10 CFR 52.47, along with the creation of a new rule, 10 CFR 50.46a (see 67 FR 50374, August 2, 2002). These proposed changes are meant to risk-inform the combustible gas control requirements, and constitute significant relaxations of the requirements. The staff plans to finalize the rule changes during 2003. DCD Tier 2 is written in anticipation of these rule changes. As such, it is not in compliance with the current, more-restrictive regulations. Furthermore, until the proposed rule changes are final and effective, the staff cannot know for certain if the DCD will comply with the revised rule. Therefore, the issue of containment combustible gas control, as well as GL-84-09, must remain open at this time.This is Open Item 6.2.5-1.
GL-84-12, Compliance with 10 CFR Part 61 and Implementation of Radiological Effluent TS, Attendant Process Control Program	This GL addressed the concern of the compliance with 10 CFR Part 61 and implementation of radiological effluent TS, attendant process control program. GL-84-12 has been superseded by GL-89-01, which has been incorporated into Technical Specification 5.5.3, “Radioactive Effluent Control Program,” in a manner consistent with the guidance provided in NUREG-1431, “the Applicant Standard Technical Specification.” In addition, DCD Tier 2 Section 11.4.6 references 10 CFR Part 61 for radioactive waste disposal containers and specifies a COL requirement that "The Combined License applicant will develop a process control program in compliance with 10 CFR Section 61.55 and 61.56 for wet solid waste." This is a part of COL Action Item 11.4-1. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-84-13, Technical Specifications for Snubbers	The purpose of this GL was to authorize the elimination of a table in plant specific Technical Specifications (TS) which required a list of all snubbers in the plant. The applicant improved STS eliminated this requirement. It was replaced by ANSI/ASME OM Part 4, which is now referenced in ASME Section XI for inservice testing and inspection of snubbers. As discussed under Bulletin 81-001 in this table and in Section 3.9.3.3 of this report, Westinghouse committed to ASME Section XI and ANSI/ASME OM Part 4. Therefore, GL-84-13 is not applicable to the AP1000.
GL-84-15, Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	This GL addresses the reliability of the EDG, which has been identified as being one of the main factors affecting the risk from SBO. Thus, attaining and maintaining high reliability of EDGs was a necessary input to the resolution of Unresolved Safety Issue (USI) A-44. As stated by the applicant in DCD Tier 2 Section 1.9, and Appendix 1A, about the inapplicability of RG 1.108, this GL is not applicable because the diesel generators in the AP1000 design are not safety-related and are not required for accident mitigation. The staff agrees. Based on the staff review, this GL is resolved for the AP1000 design.
GL-84-21, Long-Term, Low-Power Operation in PWRs	This GL was concerned with core peaking factors being greater than assumed in safety analyses for extended low-power operation followed by a return to full-power operation. In WCAP-15800, Rev. 1, Westinghouse stated that this GL is not applicable to the AP1000 design because it was an administrative communication to the licensees. However, during the review of the AP1000 safety analysis, the staff considered the effect of extended low power operation on core peaking factors. The safety evaluation of this issue is discussed in Chapter 15 of this report. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-84-24, Certificate of Compliance to 10 CFR 50.49, Environmental Qualification of Equipment Important to Safety	This GL required certification from the licensees that the plant environmental qualification program satisfies 10 CFR 50.49, has at least one path to safe shutdown with qualified equipment, and has all other equipment qualified, and or a justification for continued operation. The staff considered this issue not relevant to the design review for the AP1000. Based on the staff review, this GL is resolved for the AP1000 design.
GL-85-05, Inadvertent Boron Dilution Events	GL-85-05 informed each PWR licensee of the staff position resulting from the evaluation of Generic Issue 22, "Inadvertent Boron Dilution Events," and urged each licensee to assure itself that adequate protection against boron dilution events exists in its plants. In WCAP-15800, Rev. 1, the applicant refers this issue to DCD Tier 2 Section 15.4.6. The staff evaluated and discussed this issue in Chapter 15 of this report. For mitigating the consequence of this event, operator actions are required to isolate the potential unborated water from the demineralized water transfer and storage system, or CVS . The staff identifies COL Action Item 20.7-7 that the COL applicants should develop plant-specific EOPs that address the boron dilution events. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-85-06, Quality Assurance Guidance for ATWS Equipment That is Not Safety Related	Generic Letter 85-06 provides the explicit QA guidance required by 10 CFR 50.62 for the non-safety related SSCs required to mitigate an ATWS event per 10 CFR 50.62(c)(1). The NRC staff reviewed DCD Tier 2 Sections 15.8, 17.3, Table 17-1, and WCAP-15985, Revision 1, for applicant's resolution of this generic issue. In DCD Tier 2 Section 15.8, the applicant stated that the AP1000 diverse actuation system (DAS) provides the ATWS mitigation systems by tripping the turbine and actuates passive residual heat removal to provide decay heat removal. In DCD Tier 2 Section 7.7.2.11, the applicant described the DAS as a non-safety-related system that provides a diverse backup to the protection system. The staff's safety evaluation of the AP1000 ATWS mitigation features is described in Section 7.7.2, of this report. The applicant addressed quality assurance requirements for the SSCs providing ATWS mitigation under the regulatory treatment for non-safety systems (RTNSS) process described in SECY 95-132. WCAP-15985 provided the proposed resolution for the AP1000 RTNSS policy issue. WCAP-15985 states that the DAS functions and the associated non-class 1E DC and UPS system power supplies, are needed to meet the requirements of 10 CFR 50.62, and DAS needs to meet Generic Letter 85-06. However, WCAP-15985 did not include that GL 85-06 is also applicable to the non class 1E and UPS power systems that support the DAS ATWS functions. Therefore, the staff determined that the applicant should clearly state the quality assurance requirements that are applicable to the DAS and non-class 1E and UPS systems for the purposes of satisfying the requirements of GL 85-06. This issue is identified as Open Item 20.7-2. Pending resolution of Open Item 20.7-2, the NRC staff concluded that: (1) DAS and supporting power supplies are non-safety related systems that are subject to the quality assurance guidance contained in GL 85-06, and (2) the quality assurance controls specified in DCD Tier 2 Table 17-1 are applicable to SSCs required to comply with 10 CFR 50.62 and are equivalent to the quality assurance guidance contained in GL 85-06.

GL No. and Title	Staff Resolution
GL-85-13, Transmittal of NUREG-1154 Regarding the Davis-Besse Loss of Main and Auxiliary Feedwater Event	NUREG-1154 addressed the Davis-Besse loss of all feedwater event. The cause of the loss of main and auxiliary feedwater event on June 9, 1985, at Davis-Besse plant was (1) the licensee's lack of attention to detail in the care of plant equipment; (2) the licensee's history of poor performance in troubleshooting, maintenance, and testing of equipment; (3) the fact that licensee's evaluation of operating experience related to equipment did not always find the root causes of problems and corrected; and (4) the licensee's ineffective or unutilized engineering design and analysis effort to evaluate equipment problems. On the basis of the above, the staff finds that the Davis-Besse event is caused by inadequate system maintenance program. The AP1000 does not have an auxiliary feedwater system. The startup feedwater system, described in DCD Tier 2 Section 10.4.7.1, is not a safety-related system and is not relied on to provide safety-related cooling for the RCS. The passive core cooling system, including the passive residual heat removal heat exchangers are the safety-related means of providing emergency cooling for the RCS. The applicant addresses The Davis-Besse event in its plant operating and maintenance procedures for the main and startup feedwater systems. DCD Tier 2 Section 13.5 identifies the COL applicant as having the responsibility for the preparation of plant operating procedures. The staff agrees with the applicant's assessement. Based on the staff review, this GL is resolved for the AP1000 design.
GL-85-16, High Boron Concentrations	This GL encouraged licensees to reevaluate the need for high boron concentration (about 20,000 ppm boron) in the boron injection tank. In WCAP-15800, Rev. 1, Westinghouse stated that this GL is not applicable to the AP1000 design because the AP1000 design does not have a boron injection tank. The staff agrees because the design only has a coolant makeup tank with a maximum boron concentration of 3300 ppm boron. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-85-19, Reporting Requirements on Primary Coolant Iodine Spikes	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design because RCS activity reporting requirements are the responsibility of the COL applicant. The staff agrees. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-86-04, Policy Statement on Engineering Expertise on Shift.	This GL has been satisfactorily addressed by the applicant in DCD (Revision 3) Section 18.7, “Staffing,” and has been identified as a COL responsibility. Based on the staff review, this GL is resolved for the AP1000 design.
GL-86-07, Transmittal of NUREG-1190 Regarding the San Onofre Unit 1 Loss-of-Power and Water-Hammer Event	GL-86-07 transmitted incident investigation report NUREG-1190, and requested licensees to review the report for applicability to their facility. In WCAP-15800, Rev. 1, the applicant states that this GL is resolved in Generic Issue A-1, "Water Hammer," in DCD Tier 2 Subsection 1.9.4.2.2. The staff agrees. The staff evaluation of Generic Issue A-1 is discussed in Section 20.2 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-86-10, Implementation of Fire Protection Requirements	This GL provided guidance on meeting Appendix R to 10 CFR Part 50, which took precedence over GL-83-13. The staff included this GL in its review of the AP1000 design in Section 9.5.1 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-86-13, Potential Inconsistency Between Plant Safety Analyses and TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved. because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-86-15, Information Relating to Compliance With 10 CFR 50.49, "Environmental Qualification of Equipment Important to Safety for Nuclear Power Plants”	This GL provided guidance on licensee actions in cases where the environmental qualification of equipment is suspect and on then-current NRC policy with regard to enforcement of 10 CFR 50.49. This issue is not relevant to the design review for the AP1000 because it is a compliance issue. Based on the staff review, this GL is resolved for the AP1000 design.
GL-86-16, the applicant ECCS Evaluation Models	This GL concerned the need for additions and corrections to the the applicant ECCS evaluation models that contain the WREFLOOD and BART computer codes. In WCAP-15800, Rev. 1, the applicant refers to DCD Tier 2 Sections 15.0.11, Computer Codes Used” for the DBA analyses, and 6.3.5, “Limits on System Parameters.” As a result of its review of the Chapter 15 design basis transients and accidents analyses described in DCD Tier 2 Chapter 15, the staff concludes that this GL is not applicable to the AP1000 design. This is because the two computer codes referred in GL-86-16 are not included in the AP1000 design ECCS analysis. The ECCS evaluation models used for the AP1000 design are the WCOBRA/TRAC and NOTRUMP codes for large-break and small-break LOCA analyses, respectively. Based on the staff review, this GL is resolved for the AP1000 design.
GL 87-06, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	In WCAP-15800, Revision 1, the applicant states that this GL is addressed in DCD Tier 2 Tier 2 Chapter 16 TS, LCO 3.4.16, “RCS Pressure Isolation Valve (PIV) Integrity.” The staff evaluation of this issue is discussed in Section 3.9.6.2 of this report. The staff concludes that DCD Tier 2 Table 3.9-18 contains an acceptable list of PIVs, and LCO 3.4.16 in the TS contains acceptable leak testing criteria for these PIVs. Based on the review of TS and the information in DCD Tier 2 Table 3.9-18, the staff concludes that GL-87-06 is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-87-09, Sections 3.0 and 4.0 of STS on Limiting Conditions for Operation and Surveillance Requirements	The applicant stated that this generic letter is addressed in DCD Tier 2 Chapter 16 and Section 3.0. The staff evaluation is provided in Chapter 16 of this report. Therefore, the staff concludes that generic letter is resolved for the AP1000 design.
GL 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements	The applicant stated that this GL is addressed in DCD Tier 2 Section 3.6.2. This GL issued Revision 2 to BTP MEB 3-1 of SRP 3.6.2 to eliminate the guidelines for postulating arbitrary intermediate pipe ruptures. DCD Tier 2 Section 3.6.2 provides information relative to postulating pipe ruptures that is consistent with MEB 3-1, Revision 2. Based on the staff review, this GL is resolved for the AP1000 design.
GL-87-12, Loss of Residual Heat Removal While the RCS Is Partially Filled	As a result of the loss of the decay heat removal function occurring in operating plants, this GL requested licensees to provide information regarding midloop operation, and GL-88-17 provided guidance to licensees. In WCAP-15800, Rev. 1, Westinghouse stated that this GL is addressed in DCD Tier 2 Section 1.9.5.1, SECY-90-016 Issues. In DCD Tier 2 Subsection 1.9.5.1.4, the applicant discussed the AP1000 design's compliance with the issue of midloop operation. The staff resolved this issue and its evaluation is discussed in Section 19.3 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-88-02, Integrated Safety Assessment Program II (ISAP II)	Risk insights are already an integral part of the staff's AP1000 design review process as discussed in Chapter 19 of this report on severe accidents and PRA for the design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary in PWR Plant Components	This GL requested assurance that licensees had implemented a program to ensure that boric acid corrosion does not degrade the RCPB. In WCAP-15800, Revision 1, “Operational Assessment for AP1000,” the applicant indicated that this GL is not applicable to the AP1000 design and because it is the responsibility of the COL applicant. The staff agrees that this is an inspection issue and within the scope of the COL applicant. The AP1000 COL applicant will be developing a boric acid corrosion program to provide reasonable assurance of compliance with the applicable regulatory requirements. This action item is designated as COL Action Item 20.7-8.
GL-88-07, Modified Enforcement Policy Relating to 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants"	In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design reviews. Therefore, this GL is not relevant to the design review for the AP1000. Based on the staff review, this GL is resolved for the AP1000 design.
GL-88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Reactor Operations	The applicant stated that this GL is addressed in Section 1.9.4 and DCD Tier 2 Appendix 1A as it involves Issue A-47 and RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. Issue A-47 is on safety implications of control systems and is discussed in Section 20.2 of this report. Issues A-11 and 15 involve reactor vessel materials and radiation, and are discussed in Sections 20.2 and 20.3, respectively, of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-88-12, Removal of Fire Protection Requirements from TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment	The applicant stated that this GL is addressed in DCD Tier 2 Section 9.3.1. Staff resolution of Issue 43 is provided in Section 20.3 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-88-15, "Electrical Power Systems - Inadequate Control Over Design Process"	This GL informs the licensees of the various problems with electrical systems being identified with increasing frequency at nuclear power plants. This refers to the problems of onsite distribution system voltages lower than required for proper operation of safety equipment, diesel generator loading exceeding design, inadequate diesel generator response to actual loading, overloading Class 1E buses, inadequate breaker coordination, and inadequate fault current interruption capability. For GL-88-15, the applicant referred to DCD Tier 2 Section 8.3.1.1.2.1 in WCAP-15800, Revision 1. The standby diesel generators are included in the investment protection short-term availability controls described in DCD Tier 2 Section 16.3 and the design reliability assurance program described in DCD Tier 2 Section 17.4. The breaker coordination and fault current interruption capability are covered by ITAAC. Based on the staff review, this GL is resolved for the AP1000 design.
GL-88-16, Removal of Cycle-specific Parameter Limits from Plant TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-88-17, Loss of Decay Heat Removal	This GL concerned loss of decay heat removal during nonpower operation. In WCAP-15800, Rev. 1, Westinghouse stated that this GL is discussed in DCD Tier 2 Section 1.9.5.1. This GL and GL-87-12 are addressed in midloop operation for SECY-90-016 issues in DCD Tier 2 Subsection 1.9.5.1.4. (The SRM to SECY-90-016 provided four additional recommendations for decay heat removal during midloop operation.) The staff resolved this issue and its evaluation is discussed in Section 19.3 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-88-19, Use of Deadly Force by Licensee Guards to Prevent Theft of Special Nuclear Material	The staff has not completed review of this item. It is a part of the Open Item 13.6-1.
GL-88-20, Individual Plant Examination for Severe-Accident Vulnerabilities	Risk insights are already an integral part of the staff's AP1000 design review process as discussed in Chapter 19 of this report on severe accidents and PRA for the design. Based on the staff review, this GL is resolved for the AP1000 design.
GL-89-01, Implementation of Programmatic and Procedural Controls for Radiological Effluent TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products	The purpose of Generic Letter 89-02 was to share with all licensees some of the elements of programs that appear to be effective in providing the capability to detect counterfeit or fraudulently marketed products and in assuring the quality of vendor products. In DCD Tier 2 Section 1.9.5.5, "Operational Experience," the applicant states: 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, "Operational Assessment for the AP1000." In WCAP-15800, "Operational Assessment for AP1000,"page 3-17, Table 3, "Generic Letters," page 3-22, Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products (3/89)," the applicant states this issue is related to procurement and is not applicable to the AP1000 design. The NRC staff agrees with the applicant that Generic Letter 89-02 is not applicable to the AP1000 design certification review since this is a procurement issue. The COL applicant is responsible for procurement issues.
GL 89-04, Guidance on Developing Acceptable Inservice Testing Programs	In WCAP-15800, Revision 1, Westinghouse stated that this GL is addressed in DCD Tier 2 Sections 3.9.6.2, 5.2.4 and 6.6. The staff's evaluation and acceptance of the AP1000 IST program was based on the information in DCD Tier 2 Section 3.9.6. The details of the staff review of the AP1000 IST program are included in Section 3.9.6 of this report. Based on the staff review, this GL is resolved for the AP1000 design.
GL-89-07, Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	The staff has not completed review of this item. It is a part of the Open Item 13.6-1.

GL No. and Title	Staff Resolution
GL-89-08, Erosion/Corrosion Induced Pipe Wall Thinning	This GL requested information on the long-term erosion/corrosion monitoring program that provided assurance that the structural integrity of all high energy carbon steel systems will be maintained. The applicant stated that this GL is a surveillance issue and that this issue is discussed in DCD Tier 2 Sections 5.4.3.4 and 10.3.6. DCD Tier 2 Section 10.1.3 indicates that this issue is the responsibility of the COL applicant and that the COL applicant would address preparation of an erosion/corrosion surveillance program using industry guidelines. The staff agrees with this assessment. Based on the staff review, this GL is resolved for the AP1000 design.
GL 89-10, Safety Related Motor Operated Valve Testing and Surveillance	WCAP-15800, Revision 1 references DCD Tier 2 Section 3.9.6.2 as the basis for resolution of this GL. As discussed in Chapter 3 of this report, the staff review of this information concludes that the commitments in DCD Tier 2 Sections 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis for resolution of GL-89-10 for the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.
GL 89-10 S1, Results of the Public Workshops	WCAP-15800, Revision 1 references DCD Tier 2 Section 3.9.6.2 as the basis for resolution of this GL. As discussed in Chapter 3 of this report, the staff review of this information concludes that the commitments in DCD Tier 2 Sections 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis for resolution of GL-89-10 for the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL 89-10 S5, Inaccuracy of Motor Operated Valve Diagnostic Equipment	WCAP-15800, Revision 1 references DCD Tier 2 Section 3.9.6.2 as the basis for resolution of this GL. As discussed in Chapter 3 of this report, the staff review of this information concludes that the commitments in DCD Tier 2 Sections 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis for resolution of GL-89-10 for the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.
GL 89-10 S6, Information on Scheduling and Grouping, and Staff Responses to Additional Public Questions	WCAP-15800, Revision 1 references DCD Tier 2 Section 3.9.6.2 as the basis for resolution of this GL. As discussed in Chapter 3 of this report, the staff review of this information concludes that the commitments in DCD Tier 2 Sections 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis for resolution of GL-89-10 for the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.
GL 89-10 S7, Consideration of Valve Mispositioning in PWRs	WCAP-15800, Revision 1 references DCD Tier 2 Section 3.9.6.2 as the basis for resolution of this GL. As discussed in Chapter 3 of this report, the staff review of this information concludes that the commitments in DCD Tier 2 Sections 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis for resolution of GL-89-10 for the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.
GL-89-13, Service Water System Problems Affecting Safety-Related Systems	This GL requested information about compliance of service water systems with certain GDC and quality assurance requirements, as test control. The applicant states in WCAP-15800 (Rev. 1) that the service water system is not used for safety-related cooling in the AP1000. Therefore, this GL is not applicable to the AP1000. The staff agrees with the applicant on its assessment and this GL is resolved for the design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-89-14, Line-item Improvements in Technical Specifications —Removal of 3.25 Limit on Extending Surveillance Intervals	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-89-15, Emergency Response Data System	DCD Tier 2 Section 13.3 ("Emergency Planning") states that emergency planning, including communication interfaces among the main control room, the technical support center and the emergency planning centers, are the responsibility of the COL applicant. The COL applicant referencing the AP1000 certified design will address emergency planning, including post 72-hour actions and its communications interface. DCD Tier 2 Section 9.5.2 ("Communication System") provides that COL applicants referencing the AP1000 certified design will address interfaces to required offsite locations. Further, the emergency response facility communication system, including the crisis management radio system, will be addressed by the COL applicant. Therefore, the staff concludes that this GL is not applicable to the AP1000 design, and that it is the responsibility of the COL applicant. It is noted, however, that Appendix E to 10 CFR Part 50, Section VI 2(a)(i) provides the selected emergency response data system (ERDS) plant parameters for PWRs. Due to the unique design of the AP1000, the plant parameters required for the ERDS will be similar, but not all inclusive. The reminder to the COL applicant, to review this GL and Appendix E of 10 CFR Part 50 to ensure that the necessary and sufficient AP1000 plant parameters are available to the ERDS. This is COL Action Item 20.7-9. Based on the staff review, this GL is resolved for the AP1000 design.
GL-90-02, Alternative Requirements for Fuel Assemblies in the Design Features Section of TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-90-09, Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. This GL involved a line-item improvement to plant TSs. For the same reasons discussed in the staff's evaluations of GL-80-099 and GL-84-013 above in this table, the staff determined that GL-90-09 is not applicable to the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.
GL-91-01, Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-91-04, Changes in TS Surveillance Intervals to Accommodate a 24-month Fuel Cycle	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was resolved. because it was no longer considered relevant to the design review. The proposed AP1000 TS contain 18-month SR frequencies for cyclic SRs that can only be done during shutdown conditions, consistent with an 18-month fuel cycle. For cyclic SRs that do not require shutdown conditions to perform, the AP1000 TS propose 24-month frequencies in anticipation of a future transition to a 24-month duration core load. The staff finds this acceptable because justifying 24-month SR frequencies, in accordance with this GL, is the responsibility of the COL applicant. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-91-05, License Commercial-Grade Procurement and Dedication Programs	The purpose of Generic Letter 91-05 was to allow licensees sufficient time to fully understand and implement guidance developed by industry to improve procurement and commercial-grade dedication programs. In DCD Tier 2 Section 1.9.5.5, “Operational Experience,” the applicant states: 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, “Operational Assessment for the AP1000.” In WCAP-15800, “Operational Assessment for AP1000,”page 3-17, Table 3, “Generic Letters,” page 3-25, Generic Letter 91-05, “License Commercial-Grade Procurement and Dedication Programs (4/91)” the applicant states that this issue is related to procurement and is not applicable to the AP1000 design. The NRC staff agrees with the applicant that Generic Letter 91-05 is not applicable to the AP1000 design certification review since this is a procurement issue. The COL applicant is responsible for procurement issues.
GL-91-07, GSI-23, “RCP Seal Failure" and its Possible Effect on SBO	GI-91-07 informed licensees of the possible effect of GSI-23, “RCP Seal Failures,” on their responses to the SBO rule. In WCAP-15800, Rev. 1, Westinghouse stated that this GL is not applicable to the AP1000 design, as discussed in DCD Tier 2 Subsections 5.1.3.3 and 1.9.4.2.3. The staff concluded that GSI-23 does not apply to the AP1000 design with canned RCPs. Therefore, this GL is not applicable to the AP1000 design, and the issue is resolved.

GL No. and Title	Staff Resolution
GL-91-08, Removal of Component Lists from TS	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-91-09, Modification of Surveillance Interval for the Electrical Protection Assemblies in Power Supplies for the Reactor Protection System	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-91-14, Emergency Telecommunications	This GL alerted reactor power plant licensees of NRC's effort to implement an upgrade to its emergency telecommunications system. NRC had identified seven essential communications functions, and requested licensees to make modifications to their facilities and procedures to ensure compliance with 10 CFR 50.47(b)(6) and 10 CFR Part 50, Appendix E, Section IV.E.9.d. DCD Tier 2 Section 13.3 ("Emergency Planning") states that emergency planning, including communication interfaces among the main control room, the technical support center and the emergency planning centers, are the responsibility of the COL applicant. The COL applicant referencing the AP1000 certified design will address emergency planning, including post 72-hour actions and its communications interface. DCD Tier 2 Section 9.5.2 ("Communication System") provides that COL applicants referencing the AP1000 certified design will address interfaces to required offsite locations. Further, the emergency response facility communication system, including the crisis management radio system, will be addressed by the COL applicant. The staff considers that this issue is outside the scope of the AP1000 design certification, and that it, therefore, will be addressed by the COL applicant. This issue is covered by DCD Tier 2 Section 13.3, and is addressed in DCD Tier 2 Section 13.3.2 as COL Action Item 13.3-1. Therefore, GL-91-14 is resolved for the AP1000 design. The reminder to the COL applicant to review these GL requirements associated with emergency telecommunications. This is COL Action Item 20.7-10.
GL 91-15, Operating Experience Feedback Report, Solenoid-Operated Valve Problems at U.S. Reactors	This generic letter informed licensees of a case study on solenoid-operated valves by AEOD. No specific action was requested. The applicant stated that this generic letter is not applicable to the AP1000 design because it involved procurement and maintenance issues, which are the responsibility of the COL applicant. The staff agrees with the applicant as stated in the resolution of Issue I.C.5 in Section 20.4 of this report. Therefore, this generic letter is not applicable to the AP1000 design. However, this is part of the COL Action Item 20.4-2.

GL No. and Title	Staff Resolution
GL-91-16, Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	The staff has not completed review of this item. It is a part of the Open Item 13.6-1.
GL- 92-01, Revision1: Reactor Vessel Structural Integrity, and GL- 92-01, Revision 1, Supplement 1: Reactor Vessel Structural Integrity	GL 92-01, Revision 1, requested that licensees provide information necessary to assess compliance with requirements regarding reactor pressure vessel integrity in view of certain concerns raised in its review of RPV integrity for the Yankee Nuclear Power Station. GL 92-01, Revision 1, Supplement 1, requested licensees to consider all data relevant to reactor pressure vessel integrity. WCAP-15800, Revision 1, "Operational Assessment for AP1000," referred to DCD Tier 2 Sections 5.3.2 and 5.3.3. These sections pertain to vessel materials and P-T limits and address the structural integrity issues contained in these GLs; however, they do not address the requests in GL 92-01, Revision 1, and GL 92-01, Revision 1, Supplement 1, for providing information regarding reactor vessel integrity. Therefore, the staff concludes that it is the responsibility of the COL applicant to provide the information requested in GL 92-01, Revision 1 and GL 92-01, Revision 1, Supplement 1. This action item is designated as COL Action Item 20.7-11.
GL-92-08, Thermo-Lag 330-1 Fire Barriers	Thermo-Lag is not used in the AP1000 design, therefore, this bulletin is not applicable to the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-93-01, Emergency Response Data System Test Program	DCD Tier 2 Section 13.3 ("Emergency Planning") states that emergency planning, including communication interfaces among the main control room, the technical support center and the emergency planning centers, are the responsibility of the COL applicant. The COL applicant referencing the AP1000 certified design will address emergency planning, including post 72-hour actions and its communications interface. DCD Tier 2 Section 9.5.2 ("Communication System") provides that COL applicants referencing the AP1000 certified design will address interfaces to required offsite locations. Further, the emergency response facility communication system, including the crisis management radio system, will be addressed by the COL applicant. Therefore, the staff concludes that this GL is not applicable to the AP1000 design, and that it is the responsibility of the COL applicant. This is COL Action Item 20.7-12.
GL-93-04, Rod Control System Failure and Withdrawal of Rod Cluster Assemblies	This GL addressed the single-failure vulnerability within the the applicant solid state rod control system that could cause inadvertent withdrawal of control rods in a sequence resulting in a power distribution not considered in the DBA. In WCAP-15800, Rev. 1, Westinghouse referred to DCD Tier 2 Subsection 3.9.4, Control Rod Drive System (CRDS)," to address this issue. WCAP-13864, Revision 1-A, "Rod Control System Evaluation Program," provided the applicant Owners Group's resolution to GL 93-04, including (1) the current order timing modification to ensure that, if failures similar to those that occurred at Salem plant are present, the control rods insert symmetrically, and (2) additional surveillance tests at the beginning of each cycle. In its letter of April 2, 2003. Westinghouse stated that the AP1000 rod control system (Described in DCD Tier 2 Section 7.7.1.2) incorporates design improvements described in WCAP-13864, Rev.1, and requires preoperational and startup testing as specified in DCD Tier 2 Section 14.2.9.1.8 and 14.2.10.1.11. COL applicant will perform additional testing during operational phase of the plant. This is COL Action Item 20.7-13. The staff concludes that GL-93-04 is resolved for AP1000 design.

GL No. and Title	Staff Resolution
GL-93-05, Line-item TS Improvements to Reduce Surveillance Requirements for Testing During Power Operation	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-93-07, Modification of the TS Administrative Control Requirements for Emergency and Security Plans	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-93-08, Relocation of TS Tables of Instrument Response Time Limits	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The staff agrees. In a letter dated September 21, 1995, the staff informed the applicant that this issue was resolved, because it was no longer considered relevant to the design review. Based on the staff review, this GL is resolved for the AP1000 design.
GL-94-01, Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators	The applicant stated in DCD Tier 2 that this GL is not applicable to the AP1000 design. The onsite AC electrical power sources, the diesel generators, are nonsafety-related in the AP1000 design, and are not required to be in TS by 10 CFR 50.36. Thus this GL does not apply to the AP1000 TS. Based on the staff review, this GL is resolved for the AP1000 design.
GL-95-03, Circumferential Cracking of Steam Generator Tubes	The applicant stated in DCD Tier 2 Section 1.9.4.2.2 that this GL is addressed by addressing Generic Issue A-3. The staff's review of Generic Issue A-3 is documented above, and Generic Issue A-3 is resolved for the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL 95-07, Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves	This GL requested all holders of operating licenses or construction permits for nuclear plants to identify all safety-related power-operated gate valves in their plants that may be susceptible to pressure locking or thermal binding, and take necessary corrective actions to ensure operability of applicable valves. For the AP1000 design certification, the staff's position is that, in the design of applicable valves, a commitment to incorporate provisions to prevent these situations from occurring is sufficient to resolve this GL. WCAP-15800, Revision 1, references DCD Tier 2 Sections 5.4.8.1.2 and 5.4.8.2 for resolution of this GL. The staff review of this information concludes that DCD Tier 2 Section 5.4.8 contains sufficient commitments to design applicable valves so that there is reasonable assurance that pressure locking and thermal binding will not occur. Based on the staff review of this information, the staff concludes that GL-95-07 is resolved for the AP1000 design.
GL-96-01, Testing of Safety-Related Logic Circuits	This GL addressed problems with the testing of safety-related logic circuits. A number of NRC regulations document the requirements to test safety-related systems to ensure that they will function as designed when called upon. The applicant addresses testing of safety-related logic circuits in DCD Tier 2 Section 7.1.2. However, the action of comparing electrical schematic drawings and logic diagrams against plant surveillance test procedures to ensure that the surveillance procedures fulfill the TS requirements are the responsibility of the COL applicant. This is COL Action Item 20.7-14. Based on the staff review, this GL is resolved for the AP1000 design.
GL-96-02, Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat	The staff has not completed review of this item. It is a part of the Open Item 13.6-1.

GL No. and Title	Staff Resolution
GL-96-04, Boraflex Degradation in Spent Fuel Pool Storage Racks	This GL requested licensees who use Boraflex as a neuton absorber in its spent fuel pool storage racks (SFPSR) to assess the capability of the Boraflex to maintain a 5-percent subcriticality margin and submit an action plan if the subcriticality margin cannot be maintained. In WCAP-15800, Rev. 1, the applicant states that this is a procurement issue, and is not within the scope for the AP1000. The staff agrees. This is COL Action Item 20.7-15. The COL applicants should address the degradation of Boraflex in the SFPSR as identified in GL-96-04, and assess the Boraflex capability to maintain a 5% subcriticality margin. Based on the staff review, this GL is resolved for the AP1000 design.
GL 96-05, Periodic Verification of Design Basis Capability of Safety-Related Motor Operated Valves	WCAP-15800, Revision 1, references DCD Tier 2 Sections 3.9.6.2 and 5.4.8.5 for resolution of this GL. As discussed in Section 3.9.6 of this report, the staff concluded that DCD Tier 2 Section 3.9.6 and DCD Tier 2 Table 3.9-16 contain commitments for the AP1000 to develop an inservice test program consistent with the recommendations in GL-89-10 and its supplements, and GL-96-05 for MOVs and power-operated valves other than MOVs to demonstrate their design-basis capability throughout the plant life. DCD Tier 2 Sections 3.9.8.4 and 5.4.8.5 contain COL applicant commitments relative to inservice testing in conformance with DCD Tier 2 Section 3.9.6 and Table 3.9-16, and insitu testing to confirm the capacity of the valve to operate under design conditions. Based on the staff's review of this information, the staff concludes that GL-96-05 is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	This GL addressed concerns associated with water hammer, two-phase flow, and thermally induced overpressurization. The GL requested that licensees evaluate systems that were found to be vulnerable to these conditions, perform operability determinations as appropriate per the guidance contained in GL 91-18, and take necessary corrective actions per TS and 10 CFR Part 50 Appendix B requirements. If corrective actions were required, the GL reminded licensees of their responsibility to ensure that systems remained operable and could continue to perform their safety functions in the interim while corrective actions were being implemented. In WCAP-15800 (Rev. 1), Westinghouse stated that DCD Tier 2 Section 6.2.2 specifies that the cooling water to the containment fan coolers is not safety-related. DCD Tier 2 Section 9.4.6 specifies that the Containment Recirculation Cooling System and its supporting sub-systems are not safety-related. GL 96-06, therefore, does not apply to this system. The applicant also stated that DCD Tier 2 Section 6.2.3.1.3 specifies that the containment penetrations are protected from overpressurization. The staff finds this response acceptable. Based on the staff review, this GL is resolved for the AP1000 design.
GL-97-01, Degradation of CRDM Nozzle and Other Vessel Closure Head Penetrations	The staff concerns addressed by GL 97-01 have been superceded by Bulletins 2001-01 and 2002-02. Please see resolution to Bulletins 2001-01 and 2002-02.
GL-97-04, Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	This GL deals with assurance of adequate net positive suction head for emergency core cooling system pumps. Since the AP1000 does not use emergency core cooling system pumps, this GL is not applicable to the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL-97-05, Steam Generator Tube Inspection Techniques	WCAP-15800, Rev. 1 states that this GL is not applicable to the AP1000 design. The staff agrees because this GL relates to steam generator tube inspections that are conducted in accordance with approved inspection procedures, and as such, are outside the scope of design certification. Based on the staff review, this GL is resolved for the AP1000 design.
GL-97-06, Degradation of Steam Generator Internals	This GL requested, in part, that licensees discuss any programs in place to detect degradation of steam generator internals including a description of the plans, scope, frequency, methods, and equipment used. In WCAP-15800, Revision 1, "Operational Assessment for AP1000," the applicant indicated that this GL is not applicable to the AP1000 design since it is a procedural issue and the tube supports are fabricated from stainless steel. The staff agrees that this is a procedural issue that will have to be addressed by the COL applicant and that the likelihood of degradation of the SG internals will be less given the AP1000 SG design; however, the design does not eliminate the potential for degradation of the steam generator internals to occur. As a result, the staff concludes that the COL applicant will need to develop a program for periodic monitoring of degradation of steam generator internals. This is COL Action Item 20.7-16.
GL 98-02, Loss of Reactor Coolant Inventory and Associated Potential Loss of Emergency Mitigation Functions While in Shutdown Condition	GL 98-02 requested PWR licensees to assess the susceptibility of their RHR and ECCS to common-cause failure as a result of RCS drain-down in a shutdown condition. In WCAP-15800, Rev. 1, Westinghouse stated that this is not applicable to AP1000 because AP1000 does not rely on pumps for emergency core cooling. The AP1000 design relies on passive safety systems for the safety functions of emergency core cooling and decay heat removal. These passive safety systems do not include pumps, but relies on natural forces, such as density differences, gravity, and stored energy, to perform their safety functions. The staff agrees that GL-98-02 is not applicable to the AP1000 design. Based on the staff review, this GL is resolved for the AP1000 design.

GL No. and Title	Staff Resolution
GL 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	This GL was issued to address the potential for degradation of the ECCS and containment spray system during accident mitigation as a result of failures of protective coatings and foreign materials in containment. The applicant has addressed the control of foreign material in DCD Tier 2 Sections 6.1.2, and 6.3.8.1. Section 6.3.8.1 states that " COL Applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages." In DCD Tier 2 Section 6.1.3.2, the applicant has addressed programmatic controls to ensure the proper procurement, application, and monitoring of safety-related coatings for the AP1000. Therefore, the staff concludes that two of the three main issues raised in this GL have been resolved. With respect to the third issue of non-safety-related or unqualified coatings, an option provided by GL 98-04 was to demonstrate that compliance exists with 10 CFR 50.46(b)(5) without quantifying the amount of unqualified coatings in containment. As the AP1000 application does not contain a limit for unqualified coatings, the staff assumes that the applicant has chosen this option. The staff reviewed this section of the DCD and documented its conclusions in Sections 6.1.2 and 6.2.1.8 of this report. On the basis of that evaluation, the staff concludes that GL 98-04 is resolved for the AP1000 design.
GL 99-002, Laboratory Testing of Nuclear-Grade Activated Charcoal	DCD Tier 2 Sections 6.4, 9.4.1, and 9.4.7 stated the AP1000 design has no safety-related filtration systems. The NRC staff agrees with the applicant that Regulatory Guide 1.52 (June 2001, Revision 3), "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and GL 99-002 (June 3, 1999), "Laboratory Testing of Nuclear-Grade Activated Charcoal," are not applicable to the AP1000. Based on the staff review, this GL is resolved for the AP1000 design.