20. GENERIC ISSUES

This chapter discusses the evaluation by the staff of the U.S. Nuclear Regulatory Committee (NRC) to determine whether the Westinghouse AP1000 design complies with the requirements of Title 10, Sections 52.47(a)(1)(ii) and (iv), of the Code of Federal Regulations (10 CFR 52.47(a)(1)(ii) and (iv)), and whether the design incorporates current operating experience. As required by 10 CFR 52.47(a)(1)(iv), the applicant for a standard design certification (DC) must propose resolutions of Unresolved Safety Issues (USIs) and medium- and high-priority Generic Safety Issues (GSIs), as defined in NUREG-0933, “A Prioritization of Generic Safety Issues.” These issues must be (1) technically relevant to the design and (2) identified in the applicable supplement to NUREG-0933 that was current 6 months before the application. In addition, 10 CFR 52.47(a)(1)(ii) requires the applicant to demonstrate compliance with any technically relevant portions of the 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 to 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 10 CFR 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 Conclusions

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

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

The applicant made its application for the AP1000 standardized plant design in the Design Control Document (DCD) Tier 2, in accordance with the provisions of 10 CFR 52.45. The staff reviewed Supplement 25 to NUREG-0933 to identify the list of issues in Appendix B to NUREG-0933, “Applicability of NUREG-0933 Issues to Operating and Future Plants,” that should be addressed to conform to 10 CFR Section 52.47(a)(1)(iv). The staff also 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 developing specific guidance for future plants.

Sections 20.2 through 20.5 of this report evaluate the issues that need to be resolved for the AP1000 design to comply with the requirements of 10 CFR 52.47(a)(1)(iv). DCD Tier 2, Section 1.9.4, includes additional issues that the applicant considers applicable to the AP1000 design. The staff also evaluated these issues.

The applicant evaluated the issues in Supplement 25 to NUREG-0933 to determine those issues technically relevant to the AP1000 design.

The staff concludes that the applicant has adequately demonstrated that the AP1000 design complies with the requirements of 10 CFR 52.47(a)(1)(iv) because 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 DC applicant to demonstrate compliance with any technically relevant parts of the TMI Action Plan requirements found in 10 CFR 50.34(f). The applicant addressed these requirements in DCD Tier 2, Section 1.9.3, and Section 20.6 of this report further discusses these requirements. Because of the overlap between the TMI Action Plan items and those from NUREG-0933 (discussed in Section 20.4 of this report), Section 20.6 lists all the relevant TMI Action Plan items in tabular form. This provides the issue designation and a reference to the appropriate issue in Section 20.4 of this report, which summarizes the evaluation of the TMI Action Plan items.

The staff concludes that the applicant has adequately demonstrated that the AP1000 design complies with the requirements of 10 CFR 52.47(a)(1)(ii) because it has addressed the relevant TMI Action Plan items found 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), dated February 15, 1991, concerning SECY-90-377, “Requirements for Design Certification Under 10 CFR Part 52,” the Commission directed the staff to ensure that the DC process preserves operating experience insights in the certified design. The applicant submitted its evaluation for the AP1000 design in the Topical Report Westinghouse Commercial Atomic Power (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 because the AP1000 design addresses generic letters (GLs) and bulletins (BLs) issued by the Commission between January 1, 1980, and January 31, 2002, 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 and in Section 1.9.4, the applicant listed the issues in Supplement 25 of NUREG-0933 that it considered relevant to the AP1000 design. The applicant also justified why it did not consider an issue to be relevant to the design. Sections 20.2 through 20.6 of this report discuss the resolution of the issues that the applicant and the staff considered relevant to the AP1000 design.

Table 20.1-1 of this report 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 relevance of each issue to the design. The relevance of the issues falls into one of the following categories. The actual designation used in Table 20.1-1 of this report is included in parentheses.

- The issue is required by 10 CFR 52.47(a)(1)(ii) or (iv) (52.47).

- The issue was selected by the applicant as being relevant in DCD Tier 2, Section 1.9.4 (W).

- The staff decided to discuss the issue as being relevant to the AP1000 (staff).

The applicant justified its decision not to consider an issue as relevant to the AP1000 design in DCD Tier 2, Table 1.9-2 and Section 1.9.4. The staff reviewed the justifications for those issues that the staff considered relevant to the design to meet the requirements of 10 CFR 52.47(a)(1)(iv). DCD Tier 2, Table 1.9-2, note f states that this screening determined that the issue was not a DC issue, but is the responsibility of the Combined License (COL) applicant. This is COL Action Item 20.1.4-1.

Table 20.1-1 USIs/GSIs in NUREG-0933 (Supplement 25) relevant to the AP1000 Design

Issue	Title of Issue and Section of This Report	Relevance
	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	ATWS	52.47/W
A-11	Reactor Vessel Materials Toughness	W
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump 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 Power Sources	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 Industrial 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	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	Ductility 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 Engineered Safety Features Systems and for 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

Issue	Title of Issue and Section of this Report	Relevance
	Section 20.3, New Generic Issues	
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	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	52.47/W
57	Effects of Fire Protection System 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 Steam Line 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 Systems LOCA at LWRs	W
106	Piping and Use of Highly 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 System Reliability	52.47/W
125.II.7	Reevaluate Provision 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 Multiplant Sites	52.47/W
135	Steam Generator and Steam Line 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 Steam Generator Tube Leakage	52.47
168	Equipment Qualification of Electric Equipment	Staff
185	Control of Recriticality following Small-Break LOCAs 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

Generic Issues

Issue	Title of Issue and Section of this Report	Relevance
	Section 20.4, Three Mile Island Action Plan Items	
I.A.1.4	Long-Term Upgrading	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	Short-Term Accident Analysis and Procedures Revision	Staff
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	52.47/W
I.C.9	Long-Term Program Plan for Upgrading of 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)	Plant Status and Postaccident Monitoring	52.47/W
I.D.5(3)	On-Line Reactor Surveillance Systems	52.47/W
I.F.1	QA List	52.47
I.F.2	Develop 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 Post Accident Access to Vital Areas and Protect Safety Equipment for Post Accident Operation	52.47/W
II.B.3	Postaccident Sampling	52.47/W
II.B.8	Rulemaking Proceedings on Degraded Core Accidents	52.47/W
II.D.1	Testing Requirements	52.47/W
II.D.3	Relief and Safety 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	Reliability of Power Supplies for Natural Circulation	52.47/W
II.E.4.1	Dedicated Penetrations	52.47/W
II.E.4.2	Isolation Dependability	52.47/W
II.E.4.4	Purging	52.47/W
II.E.5.1	Design Evaluation	W
II.E.6.1	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 Implementation of all Bulletin Items	52.47/W
II.K.1(16)	Implement Procedures That Identify PZR PORV “Open” Indications and That Direct Operators to Close Manually at “Reset” Setpoint	Staff
II.K.1(17)	Trip PZR Level Bistable So That PZR 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 Systems When FW 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 ROs and SROs	52.47
II.K.1(27)	Provide Analysis and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	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 Trips	W
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite	W
II.K.3(1)	Power	W
II.K.3(2)	Install Automatic PORV Isolation System and Perform Operational Test	52.47/W
II.K.3(5)	Report on Overall Safety Effect of PORV Isolation System	52.47/W
II.K.3(6)	Automatic Trip of Reactor Coolant Pumps	Staff
II.K.3(8)	Instrumentation to Verify Natural Circulation	Staff
	Further Staff Consideration of Need for Diverse Decay Heat Removal Method	
II.K.3(9)	Independent of SGs	W
II.K.3(18)	Proportional Integral Derivative Controller Modification	W
	Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity	
II.K.3(25)	for Some Event Sequences	52.47/W
II.K.3(28)	Effect of Loss of AC Power on Pump Seals	W
II.K.3(30)	Study and Verify Qualification of Accumulators on ADS Valves	Staff
	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR Part 50,	
III.A.1.2	Appendix K	52.47/W
III.A.3.3	Upgrade Licensee Emergency Support Facilities	52.47/W
	Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Radio	
III.D.1.1	Communication Systems	52.47/W
III.D.3.3	Primary Coolant Sources Outside the Containment Structure	52.47/W
III.D.3.4	In-Plant Radiation Monitoring	52.47/W
	Control Room Habitability	
	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 Advanced Controls and Instrumentation	52.47/W

20.2 Task Action Plan Items

This section compares the task action plan items listed in Table 20.1-1 of this report to the AP1000 design. The majority of the items were chosen either because (1) the design must comply with the requirements of 10 CFR 52.47(a)(1)(iv) or 10 CFR 50.34(f), 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 the fluid systems of 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 contains evaluation results of water hammer events, as well as recommendations and measures for water hammer prevention and mitigation.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated that the AP1000 design meets the guidance of applicable Standard Review Plan (SRP) sections in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition," that provide criteria for mitigating water hammer concerns, and in NUREG-0927. The applicant also addressed design features and system operation of the AP1000 to 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 (PXS), passive residual heat removal (PRHR) system, service water system (SWS), feedwater system, and steam lines. These features are summarized below.

The automatic depressurization system (ADS) 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 in-containment refueling water storage tank (IRWST) 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 PRHR system heat exchangers (PRHR HXs) 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 (CMTs) 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 CMT. DCD Tier 2, Section 6.3, provides additional information on the PXS.

The design and operation of the feedwater delivery system minimizes the potential for water hammer in the feedwater line. The features of the SG introduce 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 an SG water hammer. In addition, the 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 protects the turbine and turbine bypass valves from water slug damage.

Based on 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, 5.4.2.2, 5.4.6.2, 5.4.7.2, 6.3.2.5, 9.2.1.2.2, 10.4.7, Chapter 14, and Section 3B.2.3. WCAP-15799, “AP1000 Conformance with SRP Acceptance Criteria,” the staff concludes that the applicant has provided acceptable commitments for the AP1000 design to meet the water hammer-related guidelines detailed in applicable sections of the SRP and NUREG-0927.

The results from a small-break loss-of-coolant accident (SBLOCA) test performed earlier for the AP600 design by Oregon State University (OSU) indicate that rapid condensation events have the potential to cause unanticipated dynamic loads in the RCS. The staff concludes that these results are applicable to the AP1000 design. The staff’s evaluation of these test results found that the induced loads are small and inconsequential to the integrity of the components and piping. Based on its 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 the 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 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,” in 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 Westinghouse 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, “Environmental and Dynamic Effects Design Bases,” to permit the application of LBB.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated that the use of LBB criteria permits the evaluation of the dynamic effects of pipe breaks to be eliminated in the analysis of structures, systems, and components (SSCs). Westinghouse uses the term “mechanistic pipe break” to refer to “leak before break.” 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. These 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.

Section 3.6.3 of this report provides the staff’s review of this information. Therefore, Issue A-2 is resolved for the AP1000 design.

Issue A-3: Westinghouse Steam Generator Tube Integrity

As discussed in NUREG-0933, Issue A-3 addresses the staff’s concerns related to SG tube degradation. These concerns stem 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 Westinghouse, 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 faced by the pressurized-water reactor (PWR) vendors.

This issue was resolved and no new requirements were established (see 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 Actions Regarding Steam Generator Tube Integrity,” dated April 17, 1985, to provide additional details on the recommended actions contained in the draft NUREG-0844, “NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity,” which was issued in September 1988.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated 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.
- Stainless steel support plates provide support to the tubes.
- Contact between tubes and support plates is via the trifoil tube hole design, which provides a high sweeping velocity to reduce sludge accumulation in crevices.

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

- 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 COL applicant will develop the SG tube preservice inspection (PSI) and ISI programs. SG tube integrity is verified in accordance with the surveillance program discussed in DCD Tier 2, Section 5.4.15. The programs are plant-specific and are contained in the technical specifications (TS) for each plant. The staff reviewed the SG surveillance program contained in Section 5.5.4, “Steam Generator (SG) Tube Surveillance Program,” of the TS found in DCD Tier 2,Chapter 16, “Technical Specifications,” and found it to be acceptable, as discussed in Section 5.4.2.2 of this report. Each plant-specific change made by a COL applicant to this surveillance program in the TS will be reviewed by the staff for each license application referencing the AP1000 DC. The staff’s review will consider the regulatory criteria in place at the time of the COL applicant’s proposed change. This action item is designated as COL Action Item 20.2-1.

Based on the above analysis, the staff concludes that Issue A-3 is resolved for the AP1000 design.

<u>Issue A-9: Anticipated Transient Without Scram (ATWS)</u>

As discussed in NUREG-0933, Issue A-9 addresses 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 ATWS is an expected operational occurrence (e.g., 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.

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:

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

- The design must comply 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 stated that the AP1000 design complies with the requirements in 10 CFR 50.62, except the AP1000 does not have a safety-related auxiliary feedwater system (AFWS). DCD Tier 2, Sections 1.9.5.1.3 and 7.7 include a discussion of the design features to minimize the probability of an ATWS.

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

Other AP1000 design features minimize the probability of ATWS occurrence and mitigate the consequences, as discussed in DCD Section 1.9.5.1.3. For the AP1000 design with PXS, 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 (see DCP/NRC1558, March 28, 2003), the applicant provided the analysis of a complete loss of normal feedwater without reactor trip using the LOFTRAN code.

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

Issue A-11: Reactor Vessel Materials Toughness

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

As discussed in NUREG-0933, Issue A-11 addresses the issue that, because of the remote possibility that nuclear reactor pressure vessels (RPVs) designed to the American Society of Mechanical Engineers (ASME) Code might fail, 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, dated October 1982, and GL 82-26, “NUREG-0744, Revision 1, Pressure Vessel Material Fracture Toughness,” dated 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 stated that the AP1000 reactor vessel design complies with the requirements of 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” and includes features to reduce neutron fluence, enhance material toughness at low temperature, and eliminate weld seams in critical areas. DCD Tier 2, Section 5.3.2, discuss material requirements, and DCD Tier 2, Section 5.3.3, provides pressure and temperature limits.

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. Sections 5.3.2, 5.3.3, 5.3.4, and 5.3.5 of this report provide the staff’s evaluation of the vessel material properties and fracture toughness.

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. The staff designated this as Issue A-12 in NUREG-0933.

This issue was resolved and no new requirements were established (see 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 based on the fracture toughness requirements of ASME Code, Section III, Subsection NF, “Supports.”

DCD Tier 2, Section 5.4.10, describes the SG and RCP supports for the AP1000 design. The supports are designed and fabricated in accordance with ASME Code, Section III, Subsection NF. Westinghouse stated 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 the 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 addresses a concern about the substantial number of snubber malfunctions, the most frequent of which include (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 with the revision of SRP Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures,” in 1981.

The staff’s review of DCD Tier 2, Section 3.9.3.4.3, concludes that the applicant’s information is consistent with the guidelines in SRP Section 3.9.3 regarding snubber operability, and acceptably addresses the issue of snubber operability. Section 3.9.3.3 of this report includes the staff’s review of this issue. On the basis of this evaluation, the staff concludes that the guidelines in SRP Section 3.9.3 regarding snubber operability have been met, and 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 addresses concerns about 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 is comprised of 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 to consider viable alternatives for regulatory requirements to ensure that ASIs have been or will be minimized in operating 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 vulnerability and reduce ASIs.

This issue concerns 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 based on 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 ASIs, as documented in WCAP-14477, “The AP600 Adverse System Interaction Evaluation Report,” and that the conclusions of WCAP-14477 are applicable to the AP1000 because the fluid system design for the AP1000 is the same as that used in 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 of this same report, dated February 2003.

The purpose of WCAP-15992 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) Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." From the time this standard originated, the industry developed methods that were used to qualify equipment in accordance with the standard. The NRC determined that a generic approach was required 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 staff designated this as Issue A-24 in NUREG-0933. This issue was resolved with the publication of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, dated July 1981.

In DCD Tier 2, Section 1.9.4, the applicant stated 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 stated that this methodology addresses the requirements of GDC 4 and 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," as well as the guidance of Regulatory Guide (RG) 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," and IEEE Std 323-1974.

On the basis of its 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 complies with the requirements of 10 CFR 50.49. 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 addresses a review of whether non-safety-related loads should also be allowed to share Class 1E power sources. 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 of this report).

The 125-V direct current (dc) emergency lighting in the main control room (MCR) and in the remote shutdown area is non-Class 1E and is fed from a Class 1E uninterruptable 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 and temperature (P/T) limits specified in the 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. This issue was resolved with the issuance of SRP Section 5.2.2, “Overpressure Protection.” Applicants for construction permits (CPs) and operating licenses (OLs) were requested to design an overpressure protection system for LWRs following the guidance provided in SRP Section 5.2.2.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated that the AP1000 design conforms to the criteria in Branch Technical Position (BTP) Reactor Systems Branch (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 (RNS) relief valve for low-temperature overpressure protection (LTOP), as described in DCD Tier 2, Section 5.2.2. Section 5.2.2 of this report provides the staff’s evaluation of the RCS overpressure protection.

The staff concludes that the AP1000 design satisfies the requirements of BTP RSB 5-2. Therefore, Issue A-26 is resolved for the AP1000 design.

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

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

Issue A-28 of NUREG-0933 addresses the development of consistent and formalized acceptance criteria for the conversion of existing spent fuel storage pools to higher density storage racks to increase storage capacity. This issue was resolved with a letter from the NRC to licensees dated 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 stated that the AP1000 design incorporates the NRC criteria, and the heat load is evaluated for the stated spent fuel storage capacity.

Section 9.1.2 of this report discusses the staff’s evaluation of the conformance of the AP1000 spent fuel pool design to the NRC criteria. Based on 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 Industrial Sabotage

Issue A-29 addresses potential methods to reduce vulnerability to sabotage. The NRC staff concluded that existing requirements dealing with plant physical security, controlled access to vital areas, screening for reliable personnel appear to be effective. This item was resolved with no new requirements.

Westinghouse stated that the passive systems in the AP1000 design provided to mitigate the effects of potential accidents may have an inherent advantage when considering potential acts of sabotage compared to the active systems in operating plants. Also, the AP1000 design includes provisions for access control to vital areas. The provisions for security are discussed in the AP1000 Security Design Report and outlined in DCD Tier 2, Section 13.6.

The staff determined that this issue was acceptably addressed by the applicant. For further information, see resolution of Open Item 13.6-1 in Section 13.6 of this report. Therefore, the staff concludes that Issue A-29 is resolved for the AP1000 design.

Issue A-31: Residual Heat Removal (RHR) Shutdown Requirements

As discussed in NUREG-0933, Issue A-31 addresses the ability of a plant to transfer heat from the reactor to the environment after shutdown, which is an important safety function. This issue 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, as well as 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 values for the hot-standby condition. Although 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, Section 1.9.4.2.2, the applicant stated that the AP1000 design includes passive safety-related core decay heat removal (DHR) systems that establish and maintain the plant in a safe-shutdown condition following design-basis events. The applicant further stated that it is not necessary that these passive systems achieve cold shutdown as defined in RG 1.139, “Guidance for Residual Heat Removal.”

The PXS is designed to maintain plant safe-shutdown conditions indefinitely. A cold-shutdown condition is only necessary for 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. DCD Section 5.4.7 discusses the non-safety-related normal RHR system (RNS). Section 5.4.7 of this report provides the staff’s evaluation of the RNS.

The applicant stated that the passive RHR system, the capability of which is discussed in DCD Section 6.3, can achieve hot-standby conditions immediately, and can reduce the reactor coolant temperature to 215.6 °Celcius (C) (420 °Fahrenheit (F)) within 36 hours. The reactor pressure is controlled and can be reduced to 1.72 MegaPascal (MPa) (250 pounds per square inch gauge (psig)). The passive RHR system also provides a closed cooling system to maintain long-term cooling. Therefore, the AP1000 complies with GDC 34, “Residual Heat Removal,” by using a more reliable and simplified system for both hot-standby and long-term cooling modes.

Further, it is not necessary that these passive systems achieve cold shutdown, as defined by RG 1.139.

In GDC 34, the NRC requires an RHR system 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 as not to exceed the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB). The safe-shutdown condition for which the RHR system should accomplish this is not defined. The Electric Power Research Institute (EPRI) Utility Requirements Document (URD) proposes that the safe-shutdown condition be defined as 215.6 °C (420 °F) for the passive advanced light-water reactor (ALWR) designs. The staff concludes that cold shutdown is not the only safe stable shutdown condition able to maintain the fuel and RCPB 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 two analyses:

(1) a safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition such that no transients will violate the specified acceptable fuel design limits and pressure boundary design limit, and that no high-energy piping failure initiated by this condition will violate the 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 and to determine the reliability/availability missions of risk-significant systems and components as a part of the effort for RTNSS

Chapters 6 and 15 of this report discuss the performance of the passive system capability. Section 19.3 of this report discusses the RTNSS issue in terms of the availability of the RNS system during shutdown and refueling conditions. The staff found that the AP1000 design acceptably addresses both these issues and found them acceptable for AP1000 design. Therefore the staff considers Issue A-31 resolved for the AP1000 design.

Issue A-35: Adequacy of Offsite Power Systems

In GDC 17, “Electronic Power Systems,” 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. Therefore, this issue is not applicable to 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 event would also create 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, "Reactor Site Criteria." With the advent of increased and longer term storage of spent fuel, the NRC determined that a need existed 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. The staff designated this 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 stated that the AP1000 design conforms to NUREG-0612 and Section 9.1.5 of the SRP. DCD Tier 2, Section 9.1.4, describes the light-load handling systems, and DCD Tier 2, Section 9.1.5, discusses the overhead heavy-load handling systems.

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

Issue A-40: Seismic Design Criteria

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

- investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites

- investigate alternative approaches, when desirable

- quantify the overall conservatism of the design sequence

- modify the NRC criteria in the SRP, when 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 the combination of modal responses. The revisions also addressed seismic analysis of the aboveground tanks and Category 1 buried piping. The revised SRP Section 3.7 provides 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 aboveground vertical tanks and buried piping.

In DCD Tier 2, Sections 2.5.2.1 and 2.5.4.5, the applicant stated that the COL applicant referencing the AP1000 design will perform a site-specific evaluation and will 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 it is acceptable for the COL applicant to perform site-specific evaluations of seismic and geotechnical characteristics.

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. Sections 3.7.1, 3.7.2, and 3.7.3 of this report discuss the staff's review of DCD Tier 2, Section 3.7. 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

Issue A-43 concerns the availability of adequate cooling water following a LOCA when long-term recirculation from the PWR containment sump or the boiling-water reactor (BWR) emergency core 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."

Section 6.2.1.8 of this report discusses the staff's evaluation of the adequacy of the IRWST and the containment recirculation screens. On the basis of its evaluation, the staff considers this issue to be resolved for the AP1000 design because the applicant adequately addressed the sump performance concerns related to Issue A-43.

Issue A-44: Station Blackout (SBO)

Issue A-44 was resolved with the publication of 10 CFR 50.63, "Loss of All Alternating Current Power," which provides requirements that LWRs be able to withstand for a specified duration and recover from an 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 the AP1000 design does not require ac electrical power to establish or maintain a plant in safe-shutdown condition. However, the design includes two redundant, non-Class 1E diesel generators (DGs) to provide electrical power for non-safety-related active systems that provide a defense-in-depth function. The non-Class 1E DGs are identified as risk-significant in the scope of the design-reliability assurance program (D-RAP) described in DCD Tier 2, Section 17.4, and submitted to the NRC on October 3, 1996. DCD Tier 2, Table 17.4-1, "Risk Significant SSCs Under the Scope of D-RAP," lists non-Class 1E DGs as RTNSS important. Section 8.5.2.3 of this report discusses the resolution of the RTNSS issue. Therefore, Issue A-44 is resolved for the AP1000 design.

Issue A-46: Seismic Qualification of Equipment in Operating Plants

Issue A-46 addresses 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," issued on February 19, 1987.

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 states that the seismic Category I 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. Section 3.10 of this report includes the staff's review of this seismic qualification methodology. Based on its 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 to become more severe as a result of control systems failures, including power supply faults. In evaluating 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. To resolve 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 included (1) SG overfill, and (2) reactor core heat removal to cold shutdown after an SB 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 (SRs) to periodically verify the operability of the overfill protection during power operation

Generic Issues

The resolution of Issue A-47 requires that the plant 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 stated that control system failures are considered as potential initiating events for the AP1000 design. 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 applicant stated that the integrated control system for the AP1000 design obtains 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 incorporates design features, such 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, and 7.7.1.8, and Figure 7.2-1, sheet 10, the applicant addressed the 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 safety-grade signal closes the MFWS control valves and isolation valves. This is provided in the RTS logic, which is sufficiently separated from the MFWS control system. The AP1000 TS 3.3.1, “Reactor Trip System Instrumentation,” and TS 3.7.3, “Main Feedwater Isolation and Control Valves,” in DCD Tier 2, Chapter 16, adequately address the SRs to verify the operability of the SG overfill protection. Therefore, the staff concludes that Issue A-47 is resolved for the AP1000 design.

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

The AP1000 DSER regarding Issue A-48 stated the following:

> This issue remains open because DCD Tier 2 does not comply with current regulations for the control of combustible gas in containment during accidents.
>
> 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 Volume 67 of the Federal Register, p.50374 (67 FR 50374), dated August 2, 2002.) These proposed changes, which constitute significant relaxations in the requirements, are meant to make the combustible gas control requirements risk informed. The staff plans to finalize the rule changes during 2003.
>
> The applicant wrote DCD Tier 2 was 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 Open Item 6.2.5-1.

Subsequent to the publication of the DSER, the NRC revised its regulations regarding the control of combustible gas in containment. The revised regulations were published on September 16, 2003, and became effective on October 16, 2003. The NRC has extensively revised 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” made associated changes to Subsections 50.34 and 52.47, and added a new section, Subsection 50.46a, “Acceptance Criteria for Reactor Coolant System Venting Systems.” The revisions apply to current power reactor licensees, and also consolidate combustible gas control regulations for future power reactor applicants and licensees. The revised rules eliminate the requirements for hydrogen recombiners and hydrogen purge systems, and relax the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance.

The ability of the AP1000 design to comply with the hydrogen control requirements of 10 CFR 50.44 is documented in DCD Tier 2, Section 6.2.4. The staff’s evaluation of the ability of the AP1000 design to comply with the hydrogen control requirements of 10 CFR 50.44 are documented in Section 6.2.5 of this report.

Concerning equipment survivability, Westinghouse addressed this issue in DCD Tier 2, Appendix 19D and in Appendix D to the AP1000 PRA, and the staff’s evaluation is documented in Section 19.2.3.3.7 of this report.

On the basis of the staff's evaluation, as documented in Sections 6.2.5 and 19.2.3.3.7 of this report, Open Item 6.2.5-1 is closed and Issue A-48 is resolved for the AP1000 design.

Issue A-49: Pressurized Thermal Shock

Pressurized thermal shock (PTS) occurs in PWRs because unanticipated transients or design-basis postulated accidents could result in severe overcooling (thermal shock) of the RPV 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. If the vessel is pressurized, the pressure stress can compound the effects of this thermal stress.

As discussed in NUREG-0933, Issue A-49 addresses the concern that neutron irradiation of the RPV 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 the integrity of the vessel. The staff is concerned about the possibility of vessel failure as a result of a severe pressurized overcooling event, or 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 the staff established new requirements in 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events." This rule establishes screening criteria that are related to the fracture toughness of the reactor vessel. The risk from 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, the applicant stated that the AP1000 design complies with the requirements of 10 CFR 50.61. DCD Tier 2, Section 5.3.4, discusses reactor vessel integrity for the AP1000 design. Material requirements and P/T limits are discussed in DCD Tier 2, Sections 5.3.2 and 5.3.3.

The staff's 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.

<u>Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments</u>

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 (Categories B, C, and D)," dated June 1978, which addressed the lack of information related to the behavior of two-way, reinforced concrete slabs loaded dynamically in biaxial tension, flexure, and shear. This issue recognized the need to develop design requirements for concrete two-way slabs to resist loading caused by a LOCA or a high-energy line break (HELB). As described below, an acceptable resolution to this issue consists of applying 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 addresses the lack of a well-defined approach for evaluating the design of steel containment vessels subject to asymmetrical dynamic loadings that may be limited by the instability of the shell. Adequately addressing the design loads, the asymmetrical vessel configurations associated with the presence of equipment hatches, and the factor of safety in determining allowable loadings represents an acceptable resolution of this issue, as described below.

With respect to Part I of this 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 the main steam isolation valve and SG blowdown compartments) are generated on the basis of a postulated HELB. The DCD also states 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 the American Concrete Institute (ACI) ACI-349 code. The use of the 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 issue, DCD Tier 2, Section 3.8.2.4.1.1, states that the buckling evaluation under external pressure used the criteria in ASME Code, Section III, paragraph NE-3133, “Components Under External Loading.” The potential buckling under overall seismic loads was evaluated in accordance with ASME Code, Case N-284, Revision 1. Section 3.8.2 of this report discusses the staff’s evaluation and conclusions for the containment shell buckling under various loads and combined load conditions. This section identified two open items (Open Items 3.8.2.1-1 and 3.8.2.2-1). Open Items 3.8.2.1-1 and 3.8.2.2-1 in the DSER were related to this issue. Both of these open items are now resolved. The resolution of these open items are included in Sections 3.8.2.1 and 3.8.2.2, respectively, of this report. On this basis, the staff concludes that the concern of Issue B-5, Part II is resolved.

Issue B-5 is resolved for the AP1000 design.

<u>Issue B-17: Criteria for Safety-Related Operator Actions</u>

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 require the operator to take action in response to certain transients. Consequently, it becomes necessary to develop appropriate criteria for SROAs. The criteria would include a method to determine those actions that should be automated in lieu of operator actions and development of a time criterion for SROAs.

American National Standards Institute (ANSI)/ANS 58.8-1984, “Time Response Design Criteria for Nuclear Safety Related Operator Actions,” includes the review criteria for this issue. Plants should perform task analysis, simulator studies, and analysis and evaluation of operational data to assess whether the plants’ engineered safety features (ESFs) and safety-related control system designs conform to the review 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 stated 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 designed to automatically actuate, provide defense-in-depth for plant events, and preclude unnecessary actuation of the safety-related passive systems. A backup manual initiation exists for both the passive and active systems.

As described in DCD Tier 2, Chapter 15, the safety systems maintain the plant in a safe condition following design-basis events. Issue A-31, above, discusses this in more detail. 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 stated that during an SG tube rupture, no operator action is required to establish safe-shutdown conditions or prevent SG overfill. As indicated in Section 18.3 of this report, WCAP-14645, Revision 2, “Human Factors Engineering Operating Experience Review Report for the AP600 Nuclear Power Plant,” satisfactorily addresses this item. The applicant has demonstrated that WCAP-14645 is applicable to the 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 addresses the staff’s concern that individual reactor fuel rods sometimes fail during normal operations, and that 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 is to ensure that these fuel failures do not result in unacceptable releases to the public. Several problems were identified in the staff analysis to improve the predictability of fuel performance, which were then addressed in the revision to SRP Section 4.2, “Fuel System Design,” in 1981. Further, fuel manufactures have taken an active role in resolving this issue since it was identified. As a result, fuel failures are now rare and the significance of this issue has been diminished. Therefore, 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. As a result, the issue was dropped from further consideration.

In DCD Section 1.9.4.2.2, the applicant stated that the AP1000 reactor core design complies with SRP Section 4.2. DCD Tier 2, Section 4.2, discusses the fuel system design.

The staff completed its review of the AP1000 fuel assembly design described in DCD Section 4.2, which is similar to the 17x17 and 17x17 XL robust fuel assemblies. Section 4.2 of this report discusses the details of fuel design and acceptance criteria. Issue B-22 is resolved for the AP1000 design.

Issue B-29: Effectiveness of Ultimate Heat Sinks

The applicant indicated in DCD Tier 2, Table 1.9-2, that it considers Issue B-29 to be relevant to the AP1000 design; however, resolution of this issue is not necessary for the AP1000 design to meet the requirements of 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-29 addresses the staff's concerns, identified in NUREG-0471, 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 selecting weather data to define the design-basis meteorology. Issues 51, 130, and 153 in Section 20.3 of this report address this issue and the need for further improvement to the design and operation of ultimate heat sinks (UHS). More specifically, this issue concerns confirming the validity of the NRC mathematical models for predicting UHS performance and providing guidance regarding the criteria for weather record selection to define UHS design-basis meteorology. This issue was resolved by staff studies which confirmed the capabilities of the 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 1.9.4.2.2, the applicant stated 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. DCD Tier 2, Section 6.2.2, describes the passive containment cooling system.

Section 6.2.2 of this report discusses the staff's evaluation of the conformance of the AP1000 design to Section 9.2.5 of the SRP. Based on the staff's conclusions in that 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 indicated that it considers Issue B-32 to be relevant to the AP1000 design; however, resolution of this issue is not necessary for the AP1000 design to meet the requirements of 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-32 addresses the staff's concerns identified in NUREG-0471 regarding the need for additional information about the potential effects of extreme cold weather and ice buildup on the reliability of plant water supplies to confirm that the design and operation of safety-related water supplies can ensure adequate operation of safety systems. SRP Section 2.4.7, "Ice Effects," offers guidance for the review of licensee submittals regarding ice effects.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated 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 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 for Normal Ventilation Systems

As discussed in NUREG-0933, Issue B-36 addresses the staff's 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 RG 1.52, “Design, Testing, and Maintenance for Postaccident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,” Revision 3 in June 2001 (for EST ventilation filter units), and RG 1.140, “Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units for Light-Water-Cooled Nuclear Power Plants,” Revision 2 in June 2001, for normal atmosphere cleanup systems.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated 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 about the conformance of the AP1000 design with RG 1.140 found in DCD Tier 2, Appendix 1A.

The staff determined that Issue B-36 is closed for the AP1000 design because the nuclear island nonradioactive 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 compares the AP1000 design to the guidelines of 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 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 required 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 any safety-related functions to mitigate design-basis accidents.

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

Issue B-56: Diesel Reliability

Issues that result in a LOOP necessitate reliance on the onsite emergency DGs for successful accident mitigation. Improving the starting reliability of onsite emergency DGs can reduce the probability of events that could lead to a core-melt accident.

The AP1000 design does not required ac power for accident mitigation. Therefore, the AP1000 DGs are non-Class 1E and their reliability is founded on industry standards and practices.

Therefore, Issue B-56 is not applicable to the AP1000 design.

Issue B-60: Loose Parts Monitoring System

The presence of a loose object in the primary coolant system can indicate degraded reactor safety resulting from failure or deterioration of a safety-related component. As discussed in NUREG-0933, Issue B-60 addresses the need to have a loose parts detection program for early detection of loose metallic parts in the primary system. The NRC has developed hardware criteria, as well as programmatic criteria, for loose parts detection programs, as described in RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," Revision 1. All CPs and OLs reviewed after January 1, 1978, are 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 indicated that Issue B-60 was resolved with no new requirements. As described in DCD Section 4.4.6.4, the AP1000 design has a digital metal impact monitoring system (DMIMS), which conforms to RG 1.133, for monitoring the RCS for metallic loose parts. Section 4.4.4.2 of this report discusses the staff's evaluation of the AP1000 DMIMS. Based on analysis in Section 4.4.4.2 of this report, the staff concludes that Issue B-60 is resolved for the AP1000 design.

Issue B-61: Allowable ECCS Equipment Outage Periods

As discussed in NUREG-0933, Issue B-61 addresses the need to establish surveillance test intervals and allowable equipment outage periods, using analytically based criteria and methods for the TS. The present TS-allowable equipment outage intervals and test intervals were determined primarily on engineering judgment. Studies performed by the NRC on operating reactors indicate that from 30 to 80 percent of the 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 stated 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 TS, the staff considered a combination of NUREG-1431, "Standard Technical Specifications Westinghouse Plants," precedent, system design, and safety-related function.

Chapter 16 of this report presents the staff's evaluation of the AP1000 TS. On the basis of this evaluation and the above, 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. Design pressures in several systems connected to the RCPB in operating plants are considerably below the RCS operating pressure. The NRC has recommended that the 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.

Resolution of this issue for the AP1000 has been subsumed by the resolution of Issue 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 staff reviewed the control room area ventilation systems and control building layout and structures to ensure that plant operators will be 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, "Control Room," 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 SRP Section 6.4, Revision 2. (For further information on this issue, see 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 stated that the MCR for the AP1000 design is essentially leaktight. 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. DCD Tier 2, Section 6.4, discusses control room habitability.

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 MCR 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, Section 6.4.5.4, the applicant committed to conducting testing for MCR in-leakage during VES operation in accordance with ASTM E741-2000. DCD Tier 2, Section 6.4.7, states that the COL applicant will provide the testing frequency for the MCR in-leakage test. This is COL Action Item 6.4-1. The staff considers Issue B-66 to be resolved because the testing described above will ensure that the AP1000 design meets the dose limits of GDC 19.

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

Issue C-1 was developed because of concerns regarding the long-term capability of hermetically-sealed instruments and equipment which must function in postaccident environments. When safety-related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water. As discussed in the resolution of Issue A-24 above, the AP1000 equipment qualification (EQ) methodology was reviewed by the staff and found to meet applicable regulations and criteria. This methodology confirms the integrity of the 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 addresses the statistical methods used for evaluating the performance 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 evaluation models found in 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," or realistic models which statistically account for uncertainties, including the uncertainty of calculation in the adverse direction. These 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.

10 CFR Part 50, Appendix K, 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. Issue C-4 addresses the NRC's 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 use of the realistic ECCS evaluation model, in addition to the evaluation model conforming to the Appendix K requirements. This best-estimate evaluation model employs 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 Section 1.9.4.2.2, the applicant stated 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 Chapter 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 addresses the specific decay heat models for the LOCA analysis models. This issue involves 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 models included in 10 CFR Part 50, Appendix K, or the realistic models supported by applicable experimental data and including uncertainty of calculation in the adverse direction. To use the Appendix K models requires that the 1971 ANS Standard, ANS-5, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," be multiplied by a factor of 1.2, to determine 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 Section 1.9.4.2.2, the applicant stated that the large-break LOCA analyses for the AP1000 design, discussed in DCD Section 15.6.5, use the decay heat model identified in the 1979 ANSI/ANS Standard 5.1.

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 addresses the issue identified in NUREG-0471 involving the staff evaluations of vendors’ data and approaches for determining LOCA heat sources, as well as 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,” dated November 17, 1983, that the statistical combination of LOCA heat sources could be used to justify the relaxation of non-required conservatism in ECCS evaluation models.

In DCD Tier 2, Section 1.9.4.2.2, the applicant stated that the discussion of LOCA heat sources for the AP1000 design is included in DCD Section 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 addresses the effectiveness of various containment sprays to remove airborne radioactive material that could be present within the containment following a LOCA. This issue 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. Section 15.3 of this report includes the staff’s evaluation of these natural removal mechanisms (such as holdup, sedimentation, and diffusion). In an SRM, dated June 30, 1997, concerning SECY-97-044, “Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design,” the Commission approved the staff’s recommendation that the AP600 include a containment spray system or equivalent for accident management following a severe accident. DCD Tier 2, Section 6.5.2, describes the containment spray system; Section 19.2.3.3.9 of this report provides the staff’s evaluation of this system. The applicant concluded in DCD Tier 2, Section 6.5.2, that inadvertent actuation of the containment spray system was not credible. The staff evaluates this conclusion in Section 6.2.1.1 of this report.

On the basis of the staff's evaluations presented in Sections 6.2.1.1, 15.3, and 19.2.3.3.9 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 as part of a process control program to package diverse radioactive plant wastes for shallow land burial. No current criteria exist for finding solidification agents acceptable.

As stated in NUREG-0933, the Commission issued 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," which defines the licensing requirements for land disposal of radioactive waste, including Section 61.56, which addresses acceptable waste characteristics. Also, the staff developed BTP Effluent Treatment System Branch (ETSB) 11-3 to be part of SRP Section 11.4, "Solid Waste Management Systems," and to 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 1.9.4.2.2, the applicant stated that the solid radwaste system for the AP1000 design transfers, stores, and prepares spent ion exchange resins for disposal. The system also provides for the disposal of filter elements, and the 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. Based on the staff's conclusions in that section, Issue C-17 is resolved for the AP1000 design.

20.3 New Generic Issues

This section compares the new generic issues of NUREG-0933 listed in Table 20.1-1 of this report to the AP1000 design. The majority of the items were chosen either because (1) 10 CFR 52.47(a)(1)(iv) or 10 CFR 52.34(f) requires 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 addresses cracking in PWR nonprimary (i.e., secondary) piping systems as a result of stress corrosion, vibratory and thermal fatigue, and dynamic loading. Cracking in PWR nonprimary system piping could lead to a decrease of the system's functional capability and could possibly result in such situations as degraded core cooling. This issue deals with occurrences of main feedwater (MFW) line cracking in certain Westinghouse and CE 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 provides conclusions regarding system safety and recommends technical solutions to the issue.

Generic Issues

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 the ultrasonic testing (UT) procedures and requirements contained in ASME Section V and Section XI 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 Appendix VIII supplements to ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a, “Codes and Standards.” 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 stated that the design and inspection requirements for feedwater lines are included in DCD Tier 2, Section 10.4.7. Further, DCD Tier 2, Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” addresses the issue of ISI for ASME Class 2 and 3 components. Section 6.6 of this report also evaluates this issue. Both DCD Tier 2, Section 6.6 and Section 6.6 of this report, discuss 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 the 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 nil-ductility transition temperature (NDTT) of the materials. As long as the operating environment in which the materials are used has a higher temperature than the NDTT of the materials, 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. The design and fabrication of reactor vessel support structures must account for this effect.

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, the applicant stated that the supports for the AP1000 reactor vessel are designed for loading conditions and environmental factors, including neutron fluence. The material requirements include fracture toughness 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 information presented above, the staff considers the reactor vessel supports for the AP1000 design to be adequately designed to withstand the effects of radiation, and Issue 15 is resolved for the AP1000 design.

Issue 22: Inadvertent Boron Dilution Events

As discussed in NUREG-0933, Issue 22 addresses the possibility of core criticality resulting from inadvertent boron dilution events during cold-shutdown conditions. 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 meet the criteria regarding fuel damage and system pressure, and terminate the dilution transient 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 the alarm announcing an unplanned dilution and the loss of shutdown margin:

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

In DCD Tier 2, Section 15.4.6, the applicant provided a safety analysis for the AP1000 design 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.

The staff identified the following two additional boron dilution scenarios in which a deborated water slug may accumulate in the RCS, and a restart of the RCPs may cause this slug to pass through the core, resulting in criticality or a power excursion:

(1) The first scenario occurs during a plant startup when the reactor is deborated as part of startup procedures. A LOOP 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. With recovery of the offsite power, the RCP restart will cause this deborated water to pass through the core.

(2) The second scenario relates to transients or accidents, such as an SBLOCA reflux condensation, during the post RCP-trip natural circulation phase, that may result in an accumulation of deborated water in the RCS loop. An inadvertent restart of the RCPs will cause this water to pass through the core.

Sections 15.2.4.6 and 15.2.8 of this report documents the staff's review of inadvertent boron dilution issues. The staff considers Issue 22 resolved for the AP1000 design.

Issue 23: Reactor Coolant Pump Seal Failures

As discussed in NUREG-0933, Issue 23 addresses the concerns about RCP seal failures that could cause an SBLOCA. The PRA analyses have indicated that the overall probability of core damage as a result of an SBLOCA 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 designing the RCP seal to ensure its integrity following an SBO for an extended period.

Generic Issues

In DCD Tier 2, Section 1.9.4.2.3, the applicant stated 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 stated that the shaft for the pump impeller and rotor is within this vessel; therefore, seals are not required. DCD Tier 2, Section 5.4.1, further discusses the canned motor pump design. The applicant concluded 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.

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 a medium-safety priority, but had not been generically resolved.

In DCD Tier 2, Section 1.9.4.2.3, the applicant stated that the AP1000 does not switch from injection to recirculation in the sense that injection is not isolated when recirculation is opened. The AP1000 provides for automatic opening of the recirculation line upon a low level signal from the 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 differs from conventional PWRs in which the flow path from the refueling water storage tank would be closed when the recirculation mode is entered. Therefore, the AP1000 design is not analogous to the design of operating PWRs in terms of Issue 24. Thus, Issue 24 is not applicable to the AP1000 design.

Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

As discussed in NUREG-0933, Issue 29 addresses the staff's concerns about the number of events involving the degradation of threaded fasteners (e.g., bolt cracking, corrosion, and failure) in operating plants from 1964 to the early 1980s. Many of the events were related to the components of the RCPB and the 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. The 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," issued April 1988, and NP-5067 "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel," Volume 1, "Large Bolt Manual," issued in 1987, and Volume 2, "Small Bolts and Threaded Fasteners," issued in 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, the applicant stated that the elements of resolving this issue pertain to operational and maintenance practices, which will be addressed by the COL applicant. It also stated 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 the applicant has adequately addressed this issue. Therefore, Issue 29 is resolved for the AP1000 design.

Issue 43: Reliability of Air Systems

Resolution of Issue 43, as discussed in NUREG-0933, is not required for the AP1000 design to meet the requirements of 10 CFR 52.47(a)(1)(ii) or (iv); however, the staff believes it should be addressed for the AP1000 design because the issue deals with all causes of air system unavailability. The issue addresses the incident at Rancho Seco in which desiccant particles in the valve operator caused the slow closure of a containment isolation valve (CIV). 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 LWRs in the United States rely on 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 on 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 resulting from degraded or malfunctioning air systems. These failures contradict the requirement that safety-related equipment dependent on air systems will either "fail safe" upon loss of air or 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 in the two volumes of NUREG-1275, “Operating Experience Feedback Report—Air Systems Problems,” 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 stated that in accordance with NUREG-1275, instrument air quality meets the manufacturer’s standards for the pneumatic equipment supplied as part of the plant. In addition, periodic checks are made to assure high-quality instrument air, as specified in ANSI/Instrument Society of America (ISA)-S7.3-1981, “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 stated that the COL applicant will address DCD Tier 2, 1.9.4.2.3, Issue 43 as part of its training and procedures identified in DCD Tier 2, Section 13.5. This is COL Action Item 9.3.1-1.

- 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 stated that the COL applicant will address DCD Tier 2, 1.9.4.2.3, Issue 43, as part of its training and procedures identified in DCD Tier 2, Section 13.5. This is COL Action Item 9.3.1-1.

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

  In DCD Tier 2, Section 9.3.1, the applicant stated that no safety-related air-operated valves (AOVs) 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 the final safety analysis report analysis.

  In DCD Tier 2, Section 9.3.1.4, the applicant stated that during initial plant testing before reactor startup, the safety systems utilizing instrument air will be tested, as part of the safety system test, to verify fail-safe operation of the AOVs 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 design. Therefore, the staff finds Issue 43 to be resolved for the AP1000 design.

Issue 45: Inoperability of Instruments Due to Extreme Cold Weather

As discussed in NUREG-0933, Issue 45 addresses 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. If these sensing lines should 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 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; to Section 7.1, Appendix A, Revision 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 stated that the AP1000 design complies with SRP Sections 7.1; Section 7.1, Appendix A; Section 7.5, “Information Systems Important to Safety”; and Section 7.7. DCD Tier 2, Appendix 1A, addresses the conformance of the AP1000 design to RG 1.151.

On this basis, 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: Proposed Requirements for Improving the Reliability of Open-Cycle Service Water Systems

As discussed in NUREG-0933, Issue 51 addresses fouling of safety-related, open-cycle SWSs 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 originally addressed only aquatic bivalves. However, the issues of flow blockage in essential equipment caused by corbicula (Issue 32) and SWS flow blockage caused by blue mussels (Issue 52) were incorporated into this issue, and Issue 51 was expanded to consider whether the NRC staff should develop new requirements for improving the reliability of open-cycle water systems. New requirements for baseline fouling programs for nuclear power plants were issued in GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” dated July 18, 1989.

Generic Issues

In DCD Tier 2, Section 1.9.4.2.3, the applicant stated that the SWS for the AP1000 design provides cooling water to the component cooling water (CCW) system and has no safety-related functions. 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 UHS is accomplished by heat transfer through the containment shell to the air and water flowing on the outside of the shell.

DCD Tier 2, Section 9.2.1, discusses the design of the SWS and the provisions for minimizing long-term corrosion and organic fouling.

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

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

Issue 57, as well as NUREG-5580, "Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety Related Equipment," addresses 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 (e.g., 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 CIV 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 addresses 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) nonredundant 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 NUREG-0737, "Clarification of TMI Action Plan Requirements," Supplement 1 dated 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, "Issue 67.3.3: AP1000 Response," the applicant stated that it followed the guidance of RG 1.97 in determining the appropriate monitoring parameters for the AP1000 design. Section 7.5 of this report describes the postaccident monitoring system.

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

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

Power-operated relief valves (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 is needed to safely shut down a plant or to 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). Issue 70, addresses 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 stated 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 levels 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.

Issue 73: Detached Thermal Sleeves

As discussed in NUREG-0933, Issue 73 addresses the staff's concerns about reports of fatigue failures of thermal sleeve assemblies in the piping systems of both PWRs and BWRs between 1978 and 1980. Five generations (0 through 4) of thermal sleeves have been used in the applicant's reactors. Only "Generation 3" thermal sleeves are 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's plants that used Generation 3 thermal sleeves. This issue was resolved for the applicant's plants with the publication of NUREG/CR-6010, "History and Current Status of Generation 3 Thermal Sleeves in Westinghouse Nuclear Power Plants," in July 1992.

In DCD Tier 2, Section 1.9.4.2.3, the applicant stated that the AP1000 does not use Generation 3 thermal sleeves. Based on the staff's review of this information, 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 addresses 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 upon 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 addressed 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 addressed 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 addressed the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

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

(4) RTS reliability improvements—This action ensures 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.

DCD Tier 2, Section 7.1, outlines the AP1000 design of the reactor trip breakers and the RPS. DCD Tier 2, Section 7.2 details information on the functional requirements for reactor trip and conformance with industry and regulatory guidance. DCD Tier 2, Sections 7.1.2.6 and 7.1.2.13 outline the provisions provided to display and record parameters used by the RTS. DCD Tier 2, Section 7.5, also provides information on the requirements for safety-related display information. Based on the staff's review of this information, 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 addresses 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 ASME Code, Section III, when added to the stresses already considered (e.g., 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 an NCC analysis conducted by an NRC contractor. Both of these analyses were performed for the B&W 177 fuel assembly reactor vessel. NUREG-1374, "An Evaluation of PWR Reactor Vessel Thermal Stress During NCC," dated May 1991, and GL 92-02, "Resolution of Generic Issue 79, 'Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown'," dated March 1992, documents the resolution of Issue 79. On the basis of conservative analyses and qualitative extrapolation of the results, the staff concluded the following in NUREG-1374:

- The B&W 177 was 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 37.8 °C (100 °F) per hour during the NCC event. The contractor used this profile 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 37.8 °F (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 licensee actions (if any), and the extent of any required following certain NCC events that may place a reactor vessel in an unanalyzed condition or outside its documented design basis.

DCD Tier 2, Section 1.9.4.2.3, references DCD Tier 2, Section 3.9.1.1.2.11, and states that the COL applicant will respond to the issues raised in GL 92-02. This is COL Action Item 20.7.1-1. 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 presented 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. DCD Tier 2, Section 3.9.1.1.2.11, specifies the total number of NCC transients used in the reactor vessel design for its 60-year life span. In addition, DCD Tier 2, Figure 5.3-3, provides a generic curve presenting operating temperature, pressure, and cooldown rate (not exceeding 37.8 °C/hr (100 °F/hr)) for the reactor vessel, which is consistent with recommendations stated in GL 92-02 and NUREG-1374. On the basis of the above information, the staff concludes that the AP1000 analyses to account for NCC events are bounded by the analyses discussed in NUREG-1374 and, therefore, are acceptable.

Thus, the staff concludes that Issue 79 is resolved for the AP1000 design.

<u>Issue 82: Beyond-Design-Basis Accidents in Spent Fuel Pools</u>

WASH-1400, (NUREG-75/014) "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975 examines the risks of beyond-design-bases accidents in the spent fuel storage pool. The report concludes that these risks were orders of magnitude below those involving the reactor core. Issue 82 in NUREG-0933 reexamines 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 suggest 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, this issue was resolved and no new requirements were established because of the large inherent safety margins in the design and construction of spent fuel pools.

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. In addition, the AP1000 includes provisions to supply water to the pool in the event the water covering the spent fuel begins to boil off.

The NRC staff reviewed the information provided by the applicant, as well as the information provided in DCD Tier 2, Section 9.1. As a result of its review, the staff considers Issue 82 to be resolved for the AP1000 design.

Issue 83: Control Room Habitability

As discussed in NUREG-0933, Issue 83 addresses 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 regarding the criteria for air filtration and adsorption units for atmospheric cleanup systems, (2) Issue B-66 regarding the control room infiltration measurements, and (3) Issue III.D.3.4, also concerning control room habitability. Sections 20.2 and 20.4 of this report discuss these three issues.

In DCD Tier 2, Section 1.9.4.2.3, the applicant stated that habitability of the MCR during normal operation is provided by the non-safety-related nuclear island VBS. In the event of a DBA 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 safety-related VES is actuated.

The safety-related VES supplies breathable quality air for the MCR operators while the MCR is isolated. In the event of an 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. In addition, a 6-hour supply of self-contained portable breathing equipment is stored inside the MCR pressure boundary.

In a letter dated May 21, 2003, the applicant committed to conform to the guidance of RG 1.78, Revision 1, to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19. In addition, the applicant agreed to revise the DCD to refer to RG 1.78, Revision 1. Confirmatory Item 6.4-1 in the DSER identified the need for inclusion of this information in the DCD. The staff has reviewed the DCD and concludes that it appropriately refers to RG 1.78, Revision 1. Therefore, Confirmatory Item 6.4-1 is resolved.

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 I, Class 1E toxic gas monitoring, as required. It also states that RG 1.78, Revision 1, addresses control room protection for toxic chemicals and evaluation of offsite toxic releases (including the potential for toxic releases beyond 72 hours) in order to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19. This is COL Action Item 6.4-1.

DCD Tier 2, Section 6.4.7, states that the COL applicant is responsible for verifying that procedures and training for control room habitability are consistent with the intent of GSI 83. This is COL Action Item 6.4-2.

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. DCD Tier 2, Section 15.6.5.3, detailed the analysis assumptions for modeling the doses to MCR personnel. Open Item 6.4-1 in the DSER, identified that the staff could not complete its review and independent dose assessment until it had resolved questions on the assumed aerosol removal rates in the containment.

In Section 6.4 of this report, the staff resolved Open Item 6.4-1 and found that the VES, under “high-high” radiation conditions as described in DCD Tier 2, Section 6.4, is capable of mitigating the dose in MCR following DBAs to meet the dose criteria specified in GDC 19 as applied to the AP1000 design.

Therefore, Issue 83 is resolved for the AP1000 design.

Issue 87: Failure of High-Pressure Coolant Injection (HPCI) Steamline without Isolation

Issue 87 addresses the staff’s 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, “Safety-Related Motor-Operated Valve Testing and Surveillance,” and its supplements on safety-related motor-operated valve (MOV) testing, GL 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Power Operated Valves,” and SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors (ALWR) Designs,” which recommended these valves be periodically tested inservice, under full flow and actual plant conditions, where practical. Furthermore, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” and SECY-95-135, “Changes to Performance Indicator Programs,” provide additional guidelines for testing MOVs.

Generic Issues

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

Issue 89: Stiff Pipe Clamps

Issue 89 addresses the staff's concerns about the use of structurally stiff clamps to support safety-related piping systems. Stiff pipe clamp designs differ 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 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 to determine the significance of any localized stresses induced in the piping. The value/impact assessment included in NUREG-0933 assigns this issue a low-priority ranking for the group of operating plants considered. However, for future plants, only the value/impact assessment results in a medium-priority ranking.

The staff's 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 using 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 staff reviewed the Westinghouse pipe support design criteria document. Based on its evaluation of this information, the staff concludes that Issue 89 is resolved for the AP1000 design.

Issue 93: Steam Binding of Auxiliary Feedwater Pumps

As discussed in NUREG-0933, Issue 93 addresses the potential for a common-mode failure of the AFWS or the emergency feedwater system (EFWS) resulting from steam binding of the AFWS pumps caused by heated 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 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 to resolve this issue. This generic 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 stated that the AP1000 design does not have a safety-related AFWS. The PXS provides the safety-related function of cooling the RCS in the event of loss of feedwater. The startup feedwater system (SUFWS) provides 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 concludes that steam binding is not a problem for the AP1000 design because the PXS does not include 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.

<u>Issue 94: Additional Low-Temperature Overpressure Protection for Light-Water Reactors</u>

As discussed in NUREG-0933, Issue 94 addresses low-temperature overpressurization events, along 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 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. Issue 94 was resolved by requiring the TS for overpressure protection to be 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 Tier 2, Section 1.9.4.2.3, the applicant stated (1) that the reactor vessel for the AP1000 was designed to be less susceptible to brittle fracture during an LTOP event, (2) that the material requirements and welding processes were developed to enhance resistance to embrittlement, and (3) that the fracture toughness of the reactor vessel is discussed in DCD Tier 2, Section 5.3.2.

As discussed in DCD Tier 2, Sections 1.9.4.2.3 and 5.4.7, one of the safety-related functions of the normal 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 to within the limits specified in 10 CFR Part 50, Appendix G. 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 I components. DCD Tier 2, Tables 3.2-1 and 3.9-16 identify these components as being subject to ISI and testing in accordance with the requirements of ASME Code, Section XI.

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

GL 90-06 addresses the establishment of additional guidance for RCS LTOP to ensure reactor vessel and RCS integrity beyond that identified in the resolution for Issue A-26. As a resolution for Issue 94, GL 90-06 requires a revision to the plant's TS concerning the capability of the LTOP system. Other possible solutions identified in GL 90-06 include hardware modifications, such as 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, "Low-Temperature Overpressure Protection (LTOP) System," found in DCD Tier 2, Chapter 16, 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 60 centimeters squared (cm^2) (9.3 inches squared ($in.^2$)) 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 stated in BASES B3.4.14 that with the RCS depressurized, a vent size of 60 cm^2 (9.3 $in.^2$) 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, therefore, is acceptable.

Issue 94 is resolved for the AP1000 design.

Issue 103: Design for Probable Maximum Precipitation

As discussed in NUREG-0933, Issue 103 addresses 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 hydrometeorological reports. The issue was resolved with the revisions to SRP Sections 2.4.2 and 2.4.3 in 1989, incorporating the probable maximum precipitation (PMP) procedures and criteria contained in the latest National Weather Service publications. This was documented in Volume 54, page 31268, of the Federal Register (54 FR 31268), issued 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 stated that the PMP is a site-related parameter; the AP1000 is designed for a PMP of 49.3 cm (19.4 inches) per hour and 15.2 cm (6 inches) in a 5-minute interval, as specified in DCD Tier 2, Table 2-1. The applicant stated that the COL applicant is responsible for demonstrating 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 detailed in DCD Tier 2, Table 2-1. For cases in which a specific site characteristic is outside the DCD Tier 2, Table 2-1 parameters, the applicant stated that the COL applicant must demonstrate that the site characteristic does not exceed the capability of the AP1000 design. DCD Tier 2, Chapter 2, provides additional information on the site interface parameters.

The COL applicant must use site-specific environmental data to determine 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 that the maximum flood level for the AP1000 design specified in DCD Tier 2, Table 2-1 shall not be exceeded by the site-specific flood level. Section 2.4 of this report further discusses this issue. This is COL Action Item 2.4.1-1.

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

Issue 105: Interfacing Systems LOCA at Light-Water Reactors (LWRs)

Issue 105 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 stated its position regarding intersystem LOCA (ISLOCA) protection in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," as well as in SECY-93-087. The staff states 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 the 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. Section 3.9.3.1.5.7, "AP1000 Design Criteria for ISLOCA," of this report, 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.

Generic Issues

For all interfacing systems and components that do not meet the full RCS URS criteria, the applicant must justify why it is not practical to reduce the pressure challenge any further. The applicant must 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 three 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 deenergized

(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 Section 1.9.5.1.7, “Intersystem LOCA,” provides the applicant’s response regarding compliance of the AP1000 design with the staff’s position on ISLOCA. The applicant stated that the AP1000 design has incorporated various design features to address ISLOCA challenges. These design features result in very low AP1000 CDF for ISLOCAs as compared to currently operating nuclear power plants. The design features are primarily associated with the RNS and are discussed in Section 3 of WCAP-15993, Revision 1, “Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria,” dated March 2003, and DCD Tier 2, Section 5.4.7. WCAP-15993 provides a systematic evaluation of the AP1000 design responses of various systems interfacing the RCS to the ISLOCA challenges.

The systematic evaluation process includes (1) a review of the AP1000 piping and instrumentation drawings (P&ID) to identify those 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 the system or subsystem is inside containment, whether it meets the three criteria specified in SECY-90-016, and whether it includes other specific design features that would prevent an ISLOCA to the extent practical. 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’s evaluation of various interfacing systems and subsystems follows.

Normal Residual Heat Removal System

The portions of the RNS from the RCS to the CIVs outside containment are designed to the RCS operating pressure. The portions downstream of the CIV and upstream of the discharge line CIV are designed so that the 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 6200 kPa (900 psig). Section 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 DHR 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 (1,200–1,250 psi) without leaking. The AP1000 RNS pump mechanical seal is similar to the Davis-Besse 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 Tier 2, Section 5.4.7.2.2, the applicant discussed the AP1000 design features in the RNS specifically aimed at reducing the likelihood of an ISLOCA. 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 4385 kPa (636 psig) connected to the RNS pump suction line inside containment, which is designed to provide LTOP of the RCS to 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. MOVs inside the containment are interlocked to prevent them from opening when the RCS pressure is above the RNS operating pressure of 3100 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 liquid radwaste system (WLS) effluent holdup tanks (EHTs), 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.

The RNS design also 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 SR 3.4.16.1. Based on the above information, the staff concludes that the RNS design meets the requirements of SECY-90-016.

Chemical and Volume Control System/Makeup Systems

DCD Tier 2, Section 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, therefore, is 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 Section 3.3.3 of WCAP-15993, the applicant stated that it is not practical to design the low-pressure portions of the makeup suction piping to higher design pressure. It is not practical 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 caused by a loss of miniflow cooling. The relief valves direct any leakage from the discharge line check valves to the WLS EHTs, which are designed to handle radioactive fluids. The EHT levels are 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. A high-pressure alarm exists 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 CIVs would automatically close to terminate the ISLOCA. In addition, the purification loop inlet isolation valves would also be closed upon 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 these protection measures meet the intent of the staff's ISLOCA position, as stated in SECY-90-016.

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 (2,485 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 Section 3.2.3 of WCAP-15993, the applicant contended that it is not practical to design the low-pressure portions of the letdown line to a higher design pressure. The WLS EHTs are large atmospheric tanks and cannot be designated to a higher design pressure. In addition, the letdown line, which is routed to the degasifier package or the EHTs, and the degasifier package, which discharges directly to the WLS EHTs, cannot be designed to a higher design pressure. The CVS letdown system includes 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 upon a 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 that the CVS letdown piping meets the staff's ISLOCA position, as stated in SECY-90-016.

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 and its drainage and level indication lines, eductor supply pump seal, and demineralized water supply line. These low-pressure portions have design pressures with the URS below the RCS operating pressure. The applicant contended that it is not practical to design the low-pressure portion of the PSS to a higher design pressure because these portions are at atmospheric pressure and connect to the low-pressure demineralized water transfer and storage system (DWS). Designing the eductor water storage tank to high pressure to meet ISLOCA criteria would require the DWS to be designed for high pressure, which is not practical.

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 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 upon 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 the 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 overpressurization of the low-pressure portions of the system. Even in the unlikely event that overpressurization occurred, 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. In addition, 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, as stated in SECY-90-016.

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. Section 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 IST plan described in DCD Tier 2, Section 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

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. DCD Tier 2, Section 9.2.4 provides the design and functional details of the DWS. 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.

Conclusions

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

Issue 106 addresses 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 addresses Issue 106 and provides alternatives for prevention of, detection of, 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 1.9.4.2.3, Issue 106, the applicant specified that the AP1000 design uses small amounts of combustible gases for normal plant operations. DCD Tier 2, Section 9.3.2, discusses the plant gas system. 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 1.9.4.2.3, also specifies that the CVS hydrogen supply piping is routed through the turbine building, 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, 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 I (see DCD Tier 2, Section 6.2.4.1).

The BTP Chemical Engineering Branch (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 CMEB 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 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, that the AP1000 design complies with Section C.5.d of BTP CMEB 9.5.1.

In addition, in DCD Tier 2, Section 9.5.1, the applicant referenced 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, based on the compliance of the AP1000 with the guidance provided in BTP 9.5.1 and the applicable NFPA standard, Issue 106 is considered resolved.

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

As discussed in NUREG-0933, Issue 113 addresses the staff's concerns in 1985 that no requirements existed for dynamic qualification testing or surveillance testing of large-bore hydraulic snubbers (LBHSs) (i.e., greater than 222411 Newtons (N) (50 (1000 pounds force) (kips)) load rating). The staff was concerned about 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 caused by HELBs.

LBHSs are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events, yet also allow for 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 stated that the AP1000 plant uses significantly fewer hydraulic snubbers than do currently operating plants. It further stated 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 Code, Subsection ISTD, is referenced as an alternative requirement for IST of snubbers in 10 CFR 50.55a(b)(3)(v), this is an acceptable commitment for periodic functional testing of LBHSs, and is in accordance with applicable portions of 10 CFR 50.55a(g)(4). DCD Tier 2, Section 3.9.3.4.3, provides 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. DCD Tier 2, Section 3.9.8.3, identifies the requirements for the COL applicant information on snubber operability testing. Section 3.9.3.3 of this report discusses further the staff's evaluation of this issue. 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 Issue 113 is resolved for the AP1000 design.

Issue 120: On-Line Testability of Protection Systems

As discussed in NUREG-0933, Issue 120 addresses requirements for conducting at-power testing of safety system components without impairing plant operation. The staff raised this issue because it 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. GDC 21, "Protection System Reliability and Testability," includes the requirements for on-line testing of protection systems. These requirements apply to both the RPS and the engineered safety features actuation system (ESFAS). A protection system with two-out-of-four logic that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three 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 Std 338. Conformance to these documents ensures that the AP1000 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 a two-out-of-four logic configuration that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three logic configuration. This meets the requirements in GDC 21 for on-line testing. The AP1000's design provision for testing of the protection system conforms to the guidelines in RGs 1.22 and 1.118. The staff concludes that Issue 120 is resolved for the AP1000 design.

<u>Issue 121: Hydrogen Control for Large, Dry PWR Containments</u>

The AP1000 DSER regarding Issue 121 stated the following:

> This issue remains open because DCD Tier 2 does not comply with current regulations for the control of combustible gas in containment during accidents.
>
> The NRC has proposed major changes to 10 CFR 50.44, 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, issued 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.
>
> The applicant wrote DCD Tier 2 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 Open Item 6.2.5-1.

Subsequent to the publication of the DSER, the NRC revised its regulations regarding the control of combustible gas in containment. The revised regulations were published on September 16, 2003, and became effective on October 16, 2003. The NRC has extensively revised 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” made associated changes to Subsections 50.34 and 52.47, and added a new section, Subsection 50.46a, “Acceptance Criteria for Reactor Coolant System Venting Systems.” The revisions apply to current power reactor licensees, and also consolidate combustible gas control regulations for future power reactor applicants and licensees. The revised rules eliminate the requirements for hydrogen recombiners and hydrogen purge systems, and relax the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance.

The ability of the AP1000 design to comply with the hydrogen control requirements of 10 CFR 50.44 is documented in DCD Tier 2, Section 6.2.4. The staff’s evaluation of the ability of the AP1000 design to comply with the hydrogen control requirements of 10 CFR 50.44 are documented in Section 6.2.5 of this report.

Concerning equipment survivability, Westinghouse addressed this issue in DCD Tier 2, Appendix 19D and in Appendix D to the AP1000 PRA, and the staff's evaluation is documented in Section 19.2.3.3.7 of this report.

The AP1000 has been provided with a system for hydrogen control that meets the requirements of 10 CFR 50.44 as evaluated in Sections 6.2.5 and 19.2.3.3.7 of this report. Therefore, Open Item 6.2.5-1 is closed and Issue 121 is resolved for the AP1000 design.

Issue 122.2: Initiating Feed and Bleed

As discussed in NUREG-0933, Issue 122.2 investigated the findings of the NRC's 1985 inspection of the loss-of-feedwater event at Davis-Besse which occurred on June 9, 1985. The issue addresses 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 event, 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 the final safety evaluation report for AP600 design, the staff reported that the Westinghouse Emergency Response Guidelines (ERGs) include the feed and bleed emergency guidelines (AFR-H.1, "Response to Loss of Heat Sink"), are reviewed, and found acceptable. In Chapter 18 of this report, the staff has concluded that the Westinghouse ERGs for AP600 design are applicable to AP1000 design. Therefore, the staff concluded that Issue 122.2 regarding initiating feed and bleed is resolved for the AP1000 design.

Issue 124: Auxiliary Feedwater System Reliability

Is response to the loss-of-feedwater event at the Davis-Besse plant in 1985, Issue 124 in NUREG-0933 addresses increasing the 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 a loss of normal feedwater flow, including loss resulting from an offsite power failure. It also 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 circumstances:

- 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 an 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 stated that this issue is not applicable to the AP1000 design because the design does not have a safety-related AFWS. The PXS will provide the safety-related function of cooling the RCS in the event of loss of feedwater to the SGs. The SUFWS, which has no safety-related function beyond containment isolation, provides the SGs with feedwater during startup, hot standby, and 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. Therefore, Issue 124 is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue 125.II.7 addresses the long-term actions from NUREG-1154, “Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985,” 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 addresses 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 because of positive moderator reactivity feedwater caused by overcooling of the RCS inventory. If the EFW were not isolated, the peak containment pressure for secondary-side breaks might 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. 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 System Reliability

Issue 128 addresses the reliability of onsite electrical systems and encompasses Issues 48, 49, and A-30. The staff has reviewed the applicant's submittal and concludes 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" for the following reasons:

- Issue 48—the applicant provided the LCO in the event of a loss of one or more Class 1E 120 V ac 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 has evaluated the Class 1E dc distribution system design for the aspects addressed by Issue A-30 in Section 8.3.2.1 of this report and concludes that it is acceptable.

Therefore, Issue 128 is resolved for the AP1000 design.

Issue 130: Essential Service Water Pump Failures at Multiplant Sites

As discussed in NUREG-0933, Issue 130 addresses 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 crosstie with one of the two Unit 2 essential service water (ESW) pumps. This issue raised concerns relative to multiplant units that have only two ESW pumps per plant, but have crosstie capabilities. A limited survey of the applicant's plants helped to identify the generic applicability of vulnerabilities of multiplant configurations with only two ESW pumps per plant. In the multiplant configurations identified (approximately 16 plants), all plants can share ESW pumps via a crosstie between plants. Additional efforts to resolve this issue included (1) a limited survey of the applicant's plants to determine the generic applicability of similar multiplant configurations with two ESW pumps per plant and whether crosstie capabilities existed, (2) a survey of B&W, and Asea Brown Boveri Company-Combustion Engineering (ABB-CE) plants to identify similar multiplant configurations, and (3) a survey of single-unit plants to determine if similar ESW vulnerabilities existed.

In DCD Tier 2, Section 1.9.4.2.3, the applicant stated 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 crosstie systems or components with another plant. In DCD Tier 2, Section 3.1.1, the applicant stated that if more than one unit is built on the same site, none of the safety-related systems will be shared. This is part of COL Action Item 20.1.4-1. The staff finds this acceptable for the AP1000 design.

Therefore, Issue 130 is resolved for the AP1000 design.

Issue 135: Steam Generator 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 an 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 position 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 of Issue 135 is 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 1.9.4.2.3, 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 to the regulatory guidance except where state-of-the-art advances have enhanced ISI 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 in accordance with RG 1.83. The AP1000 SGs include features to enhance robotic inspection of tubes without manned entry. As discussed in Sections 5.2.4 and 5.4.2 of this report, the COL applicant is responsible for the development of the SG tube PSI and ISI programs. 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 are contained in Section 5.5.4, “Steam Generator (SG) Tube Surveillance Program,” of the TS found in DCD Tier 2, Chapter 16, and found it to be acceptable, as discussed in Section 5.4.2.2 of this report. Plant-specific changes made by a COL applicant to this surveillance program in the technical specifications will be reviewed by the staff individually for each license application referencing the AP1000 DC against the staff’s regulatory criteria in place at the time of its review. As discussed in Issue A-3, this is designated as COL Action Item 20.3-1.

- Task 2 on SRP Section 15.6.3, “Radiological Consequences of SG Tube Failure”—DCD Tier 2, Section 15.6.3.1.4, discusses anticipated operator recovery actions and the effects of those actions in the mitigation of an SGTR event. DCD Tier 2, Section 15.6.3.2, describes the automatic SG overfill protection, and DCD Tier 2, Section 7.2, describes the control logic.

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

  – DCD Tier 2, Section 15.6.3, discusses radiological consequences.

  – DCD Tier 2, Section 15.6.3, discusses the SGTR design basis.

- DCD Tier 2, Section 5.4.2.5 and Appendix 1A discuss the supplemental tube inspections.

- DCD Tier 2, Section 5.4.2.4.3, discusses denting criteria.

- DCD Tier 2, Section 7.5, discusses safety-related display information.

- DCD Tier 2, Section 7.3.1.2.5, discusses RCP trip.

- DCD Tier 2, Sections 7.5 and 18.8 discuss control room design and design process.

- DCD Tier 2, Section 18.9, discusses the development of EOPs.

- DCD Tier 2, Chapter 13, discusses organization responses as part of the COL application.

- DCD Tier 2, Section 7.7.1.6, discusses reactor coolant pressure control.

• Task 4 on SG overfill, carryover, and water hammer—DCD Tier 2, Section 15.6.3.2, discusses SG overfill, water carryover, and water hammer; DCD Tier 2, Section 7.2 discusses the control logic.

Therefore, the staff concludes that Issue 135 is resolved for the AP1000 design.

<u>Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits</u>

As discussed in NUREG-0933, Issue 142 addresses observations made by the staff in 1987 during evaluation tests for the safety parameter display system (SPDS). These tests revealed that for electrical transients below maximum credible levels, a relatively high level of noise could pass through certain 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 could cause the isolation device to give a false output. This issue addresses electrical isolators used to maintain electrical separation between Class 1E and non-Class 1E electrical systems and to 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.

DCD Tier 2, Sections 7.1.2.10, “Isolation Devices,” 7.7.1.11, “Diverse Actuation System,” and WCAP-15776, “Safety Criteria for the AP1000 Instrument and Control Systems,” Section 3.9, “Conformance to the Requirements to Maintain Independence Between Safety Systems and Other Interconnected Equipment (Paragraph 5.6.3.1 of IEEE Std 603-1991, “IEEE Standard

Criteria for Safety Systems for Nuclear Power Generating Stations") describe the use of isolation devices in the AP1000 I&C architecture. The isolation devices are tested to conform to design requirements. This testing will identify the devices potentially susceptible to electrical leakage. The applicant further stated that the COL license holder is responsible for implementing an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems. This is COL Action Item 20.3-2.

The use of fiber optic data links eliminates electrically conductive paths between receiving and transmitting terminals, as well as 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. DCD Tier 2, Section 7.1.2.8, "Communications Functions," describes these data links. The electromagnetic design, testing, and qualification is performed as described in WCAP-15776, Section 2.6 "Design Basis: Range of Conditions for Safety System Performance (Paragraph 4.7 of IEEE Std 603-1991).

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, as well as COL Action Item 20.3-2, the staff considers Issue 142 to be resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue 143 addresses problems with safety system components and control systems that have been experienced in recent years at several nuclear plants due to a partial or total loss of the plant's heating, ventilation, and air conditioning (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 DGs, 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 believes 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 depends 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 to locate areas in which the dependence of equipment operability on the 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 stated 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 the HVAC system 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 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 UHS. 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 FPS, 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 NSSS. As a result, generic solutions (if needed) are likely to be different for PWRs and BWRs. Possible solutions include (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 multiplant sites, discussed earlier in this section, the staff surveyed seven multiplant sites and found that the 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 multiplant 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 stated 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 Steam Generator Tube Leakage

Issue 163, "Multiple SG Tube Leakage," 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 events could lead to core damage resulting 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 DCD Tier 2, Table 1.9-2 indicates that Issue 163 is unresolved pending generic resolution. In response to a staff question (RAI 440.184, Revision 1, applicant letter DCP/NRC1566, April 7, 2003), the applicant stated that the issue should be considered closed for the AP1000 based on the following evaluation.

The AP1000 plant response to an MSLB scrams the reactor automatically and removes decay heat via the intact SG or the PRHR HX. 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 though it were an SBLOCA. The CMTs drain and produce a low-level signal. The plant protection and monitoring system depressurizes the RCS via the 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 the leaking SG tubes. Therefore, no long-term core uncovery is expected. Also, the elevation at 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 concluded 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. Issue 163 concerns the possibility that a multiple SGTR, resulting from an MSLB and degraded 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, an SGTR is mitigated using the PXS, initially through the PRHR HX, and the CMTs. After the CMTs drain to the low level to actuate the ADS, the RCS depressurization would result in gravity injection from the IRWST, and eventually from the containment recirculation. The scenario that the safety injection from the refueling water storage tank, which is outside the containment in the existing plants, will be depleted to result in core damage is not likely for the AP1000 design because the IRWST and containment recirculation will continue to provide core cooling.

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

Therefore, Issue 163 is resolved for the AP1000.

Issue 168: Equipment Qualification of Electric Equipment

This issue relates 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 it should be reassessed for the current license term. This issue is not applicable to the AP1000 design, and COL actions on the AP1000 will be based upon current cable requirements. Therefore, reassessments are not required for the AP1000. This is COL Action Item 20.1.4-1.

Therefore, Issue 168 is resolved for the AP1000.

Issue 185: Control of Recriticality Following Small-Break LOCA in PWRs

As discussed in NUREG-0933, Issue 185 addresses the possibility of a recriticality because of the potential for an unborated water slug to enter the core following an SBLOCA event. Specifically, the issue was identified following an NRC Office of 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-burnup 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-burnup fuel may be more susceptible to reactivity events than previously expected. In addition, calculations conducted by the B&WOG have predicted that prompt criticality is possible, and that significant heat generation under these conditions may result from SBLOCAs.

The applicant has addressed this issue in the context 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 SBLOCA 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 primarily concerns 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 involving an SBO. At this time, all of the operating plants that have hydrogen igniters (and 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 the staff considered the applicability of Issue 189 to AP1000.

In DCD Tier 2, Table 1.9-2, the applicant identified Issue 189 as either having a priority of “low, drop” or as not having been prioritized. The staff agrees with this assessment for the following reasons:

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

- Despite the low contribution to CDF from SBO, the AP1000 design has the capability to power the igniters from non-safety-grade DGs or station batteries in the event of a SBO.

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

<u>Issue 191: Assessment of Debris Accumulation on PWR Sump Performance</u>

Like Issue A-43 (see earlier 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 reexamine 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 this research and analysis is incomplete. Section 6.2.1.8 of this report provides the staff’s evaluation of the AP1000 suction screens in accordance with the current state of knowledge regarding Issue 191.

Therefore, Issue 191 is resolved for the AP1000.

20.4 <u>Three Mile Island Action Items</u>

<u>Issue I.A.1.4: Long-Term Upgrading</u>

As discussed in NUREG-0933, Issue I.A.1.4 addresses 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. DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue was not relevant to the AP1000 design because it is not a DC 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 DC. Section 13.1 of this report discusses the organizational structure of the site operator. The COL applicant will be responsible for addressing this issue as part of the licensing process. This is COL Action Item 20.4-1.

Therefore, Issue I.A.1.4 is resolved for the AP1000 design.

Issue I.A.2.6(1): Revise Regulatory Guide 1.8

As discussed in NUREG-0933, Item I.A.2.6(1) addresses the revision of RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," following the publication of NUREG-0737. The revisions to the RG address an 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 (see 52 FR 16007). This issue is resolved by the 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 DC. Section 13.1 of this report discusses the organizational structure of the site operator. The COL applicant will be responsible for addressing this issue as part of the licensing process. This is COL Action Item 20.4-2.

Therefore, Issue I.A.2.6(1) is resolved for the AP1000 design.

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

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

DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue was not relevant to the AP1000 design because it is not a DC 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 DC. Section 13.2 of this report discusses training materials. The COL applicant will be responsible for addressing this issue as part of the COL process. This issue is a 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 addresses the capabilities of training simulators. This issue was resolved by Revision 1 of 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. DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue was not relevant to the AP1000 design because it is not a DC issue, but is the responsibility of the COL applicant. This is part of COL Action Item 20.4-2. As indicated in Section 18.3 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: Short-Term Accident Analysis and Procedures Revision

As discussed in NUREG-0933, Issue I.C.1 addresses the preparation of emergency operating procedures (EOPs). The information in the 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 comply with the guidance provided in NUREG-0737, as well as Supplement 1 to this same report.

DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue is not an AP1000 DC issue, but is the responsibility of the COL applicant. The staff has identified in COL Action Item 18.9.4-1 that the COL applicant should develop plant-specific EOPs using the guidance provided by the ERGs.

DCD Tier 2, Section 18.9 states that WCAP-14690, "Designer's Input to Procedure Development for the AP600," provides the COL applicant information on the development and design of the AP600 ERGs and EOPs.

There are fundamental differences between the low pressure (LP) reference plant and the AP1000 design in terms of the safety system design, operation, and philosophy of emergency mitigation and recovery. Unlike the LP reference plant in which the safety systems are active systems, the safety systems in the AP1000 are passive systems. Active systems are non-safety-related systems providing defense-in-depth functions. Even though the passive safety systems perform functions similar to those performed by 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 the plant conditions at which these systems will be actuated and will remain operating, differ from the LP reference plants. For the AP1000, the active systems, though not actuated by safeguard signals, are manually actuated and relied upon as the first line of defense to avoid unnecessary actuation of passive safety systems.

The plant responses, including possible ASIs between the active and passive systems, may also differ significantly from the LP reference plants. Certain issues in which operator actions play key roles in the accident scenarios require the AP1000-specific ERGs as a basis for resolution. For example, in an SGTR event, the 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 resulting containment bypass. Additionally, the ERGs should include guidance for (1) low-power and shutdown operations, (2) times when many systems will be out for maintenance and the plant is in a configuration different from normal operation, and (3) severe accident management.

To satisfy these requirements, the staff considered the need for the ERGs and supporting analyses to demonstrate the effectiveness of operator actions in response to transients and accidents. As indicated in Section 18.3, of this report, the applicant, in WCAP-14645, has satisfactorily demonstrated the effectiveness of operator actions in response to transients and accidents. Therefore, Issue I.C.1 is resolved for the AP1000 design, based on COL Action Item 18.9.4-1.

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

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

In DCD Tier 2, Section 1.9.3, Item (3)(i), the applicant stated that the AP1000 design engineers are continually involved in reviewing industry experiences from sources such as NRC BLs, licensee event reports (LERs), NRC request for information letters to licensees, Federal Register information, and NRC GLs. The applicant further stated that it had incorporated lessons-learned experience into the AP1000 design through its participation in the development of Volume III of the ALWR URD and in the activities of the ALWR Utility Steering Committee.

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 DC and is the responsibility of the COL applicant. This is part of COL Action Item 20.4-2. Therefore, Issue I.C.5 is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue I.C.9 addresses 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.

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

However, the applicant should identify the responsibility of the COL applicant in procedure development. It should address the methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures.

This is a COL action item in DCD Tier 2 and as discussed in Section 18.3, of this report, the applicant has 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 addresses licensee performance of a detailed review of the 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, in which additional requirements were issued. This issue is considered resolved.

Generic Issues

In DCD Tier 2, Section 1.9.3, Item (2)(iii), the applicant stated that the AP1000 MCR was designed by a multidisciplined, 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, Section 1.9.1, provides information on the conformance of the design with applicable RGs.

As indicated in DCD Tier 2 and discussed in Section 18.3 of this report, 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 addresses improving the presentation of the information provided to control room operators. Supplement 1 to NUREG-0737 includes the requirements for this issue. This issue raised the need for an 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 to provide such a display to operators, and to be capable of displaying a full range of important plant parameters and data trends on demand, thus indicating when process limits are being approached or exceeded.

In DCD Tier 2, Section 1.9.3, Item (2)(iv), the applicant stated 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.

DCD Tier 2, Chapter 18 discusses the SPDS design. The SPDS requirements are specified during the MCR design process, as 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 are similarly grouped, facilitating an easy transition for the operators. The SPDS requirement for presenting 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 the technical support center (TSC).

As indicated in Section 18.3, 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 Monitoring

As discussed in NUREG-0933, Issue I.D.3 addresses 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 stated that the AP1000 MCR meets the NRC RG 1.47 recommendations, including the 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 I&C 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 these data is incorporated in the process data displays on the graphic CRTs located in the AP1000 MCR.

The AP1000 design incorporates this information into the alarm system, the operator's workstation, and the wall panel information system in the MCR. 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. The alarm system continuously monitors this information.

Based on the above information, the staff concludes that the AP1000 design meets the guidelines of 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. Issue I.D.3 is resolved for the AP1000 design.

Issue I.D.5(2): Plant Status and Postaccident Monitoring

As discussed in NUREG-0933, Issue I.D.5(2) addresses 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 10 CFR Part 50, Appendix B (Criterion XIV) 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 include the following:

- 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 MCR and should have manual input capability.

- For postaccident 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 stated that the AP1000 design conforms to and meets the intent of RG 1.97, which provides acceptable guidance for postaccident 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 to 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. DCD Tier 2, Section 7.5, further discusses this analysis.

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 compared the AP1000 design against the criteria in RG 1.97, Revision 3, and addressed accident monitoring instrumentation. The applicant concluded that the AP1000 design complies with RG 1.97, Revision 3.

Issue I.D.5(2) was resolved with the issuance of RG 1.97, Revision 2. On the basis of the information provided by the applicant and the fact that the AP1000 design is in compliance with RG 1.97, Revision 3, 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): Online Reactor Surveillance Systems

As discussed in NUREG-0933, Issue I.D.5(3) addresses the benefit to plant safety and operations of continuous, online 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, thus allowing them to predict and identify 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. Online surveillance data have been used to assess loose thermal shields.

In DCD Tier 2, Section 1.9.4.2.1, the applicant stated that the AP1000 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 GDC 30, “Quality of Reactor Coolant Pressure Boundary.” The applicant also stated that a digital metal impact monitoring system (DMIMS) monitors the RCS for the presence of loose metallic parts, and that this system conforms with the guidance provided in RG 1.133, Revision 1.

The acceptance criteria for leak monitoring are in RG 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” 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 offers 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 a detailed discussion of the detection of leakage through the RCPB. Section 5.2.5 of this report describes the staff’s 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 4.4.4.2 of this report, the staff concludes that Issue I.D.5(3) is resolved for the AP1000 design.

Issue I.F.1: Expanded Quality Assurance List

As required by 10 CFR 52.47(a)(ii), an applicant for DC must demonstrate compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f). As required by 10 CFR 50.34(f)(3)(ii), an application must provide sufficient information to “ensure that the 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 QA controls.

The NRC staff reviewed the QA controls described in the AP1000 DC application that are applicable to the non-safety-related SSCs to verify that adequate controls were specified to ensure the reliability and availability of risk-significant, non-safety-related SSCs. The staff determined that quality programs, such as the RAP and the RTNSS, are sufficient to provide reasonable assurance that non-safety-related SSCs that are important to safety will perform satisfactorily in service. Section 17.4 of this report discusses the staff evaluation of the AP1000 RAP; Section 17.3 and Chapter 22 of this report discuss the staff’s evaluation of the RTNSS program.

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 concludes that the requirements of 10 CFR 50.34(f)(3)(ii) are not technically relevant to the AP1000 DC. Therefore, the requirement to include all SSCs important to safety in the QA list is not applicable to the AP1000 DC.

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

As required by 10 CFR 52.47(a)(ii), an applicant for DC must demonstrate compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f). As stated in 10 CFR 50.34(f)(3)(iii), an application must 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 DC 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 DC applicant. Similarly, the requirements of 10 CFR 50.34(f)(3)(iii)(G) were associated with control of “as-built” documentation and, therefore, were not technically relevant to DC. In DCD Tier 2, Section 1.9.3, the applicant indicated that the AP1000 QA plan described in DCD Tier 2, Section 17, meets the requirements of 10 CFR 50.34(f)(3)(iii).

As required by 10 CFR 52.47(a)(iv), an application for DC must contain proposed technical resolutions of those medium- and high-priority GSIs identified in the version of NUREG-0933, current on the date 6 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 include the following:

- 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
- 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 DC application. The staff concluded that because Items I.F.2(2), (3), (6), and (9) were resolved by a revision to SRP Chapter 17, a review of the QA program conducted in accordance with 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 Part 50, Appendix B. In addition, the NRC staff planned to conduct an inspection of the implementation of the quality plan to verify that design activities conducted for the AP1000 project comply with the applicant's QA program and the requirements of 10 CFR Part 50. The NRC staff planned 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 was identified as Open Item 17.3.2-2. As discussed in Section 17.3.2 of this report, the inspection teams determined that Westinghouse had adequately implemented their QA program. Therefore, Open Item 17.3.2-2 was resolved. Therefore, Issue I.F.2 is resolved for the AP1000 design.

Issue I.G.1: Training Requirements

This TMI action plan item calls for a new OL 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 OL 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 addressed this item. The applicant addresses the overall preoperational and start-up testing program in Chapter 14, "Initial Test Program." With regard to initial test program training, the applicant indicated, for example, that the results of performing natural circulation testing will be used as input into operator training. The applicant further stated that data obtained from the first-plant-only natural circulation tests using SGs and PRHR are 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 stated that the COL applicant is responsible for developing an operator training program. In addition, Sections 14.2.5, "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," of this report state that the NRC staff will defer review of the trial use of 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

As required by 10 CFR 52.47(a)(iv), an application for DC must contain proposed technical resolutions of those medium- and high-priority GSIs identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date 6 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.

Generic Issues

In DCD Tier 2, Section 1.9.4.2.1, the applicant stated that the program plan for preoperational and start-up testing of the AP1000 design is in DCD Tier 2, Section 14.2, “Specific Information to be Included in Standard Safety Analysis Reports,” which addressed SRP Section 14.

The NRC staff has concluded that the initial test program conducted in accordance with SRP Section 14.2 is adequate. Section 14.2 of this report documents the staff’s evaluation of the test program and the test program scope. Therefore, Issue I.G.2 is resolved for the AP1000 design.

<u>Issue II.B.1: Reactor Coolant System Vents</u>

As discussed in NUREG-0933, Issue II.B.1 addresses 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, “Seismic Design Classification,” Revision 3, September 1978)

In DCD Tier 2, Section 1.9.3, Item (2)(vi), the applicant stated that the AP1000 design includes the capability for remotely venting the high points of the RCS using the safety-related ADS valves and the safety-related reactor vessel head vent (RVHV) system. Both discharge to the IRWST. The ADS provides redundant groups of MOVs connected to the top of the pressurizer and squib valves connected to the top of each RCS hot leg. However, only the pressurizer MOVs (i.e., the first-stage ADS valves) are used for remote manual venting because they are the only ADS valves capable of being throttled. The RVHV system removes steam and noncondensable gases directly from the reactor vessel head (RVH).

The applicant further stated that during normal and moderate frequency events, noncondensable gases from the RCS accumulate in the pressurizer steam space with very little accumulation in the RVH 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 PXS may operate and effectively deliver cooling flow to the core. This would not happen until the RCS pressure was brought down to the operating level of the PXS.

The applicant also stated that it is possible that continued depressurization of the RCS by the ADS could create a gas-steam volume (bubble) in the upper region (head) of the reactor vessel. With only the ADS in operation, 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 the reactor vessel and pressurizer needing to be refilled.

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

The RVHV 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 RVHV system isolation valves also provide containment isolation.

The system has the capability to remove noncondensable gases or steam from the RCS using 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's evaluation of the RVHV system and the ADS design, respectively. The staff concludes that the AP1000 design complies with the requirements of 10 CFR 50.34(f)(2)(vi); therefore, Issue II.B.1 is resolved for the AP1000 design.

<u>Issue II.B.2: Plant Shielding To Provide Access to Vital Areas and Protect Safety Equipment for Postaccident Operation</u>

As discussed in NUREG-0933, Issue II.B.2 addresses licensee performance of 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 MCR, 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 postaccident operations of these systems. This issue was resolved and the requirements were provided in 10 CFR 50.34(f)(2)(vii).

In DCD Tier 2, Section 1.9.3, Item (2)(vii), the applicant stated 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 stated that improvements were incorporated into the AP1000 shielding design as they were identified.

DCD Tier 2, Section 12.2, addresses postaccident radiation sources used in the shield design and assessment of postaccident access to vital areas. The postaccident source term used for the AP1000 is predicated on the core release model from NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” which supersedes the TID-14844 source term assumptions 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. DCD Tier 2, Section 12.3, addresses vital areas for postaccident access and includes radiation zone maps that show projected dose rates in these areas and access routes for the various postaccident 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 postaccident accessibility. For each of these areas, the applicant performed an analysis to determine the dose to the individuals performing these postaccident actions. These analyses, which utilized the appropriate time-dependent postaccident dose rates and the required postaccident 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, based on the information presented above, Issue II.B.2 is resolved for the AP1000 design.

<u>Issue II.B.3: Postaccident Sampling</u>

The requirements for the postaccident sampling system (PASS) can be found 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 PASS requirements from the TS for operating plants (see 65 FR 65018, dated October 31, 2000).

As discussed in NUREG-0933, Issue II.B.3 addresses upgrading postaccident sampling at plants, the requirements for which can be found in 10 CFR 50.34(f)(2)(viii). Issue II.B.3 specifically addresses a licensee’s radiological and chemical sampling and analysis capability under transient or accident conditions, including related radiation exposures. The purpose of the PASS is to (1) provide sources of information for use by decisionmakers in developing protective action recommendations, and (2) assess core damage.

In DCD Tier 2, Section 1.9.3, Item (2)(viii), the applicant stated 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 approach includes contingency plans to obtain and analyze highly radioactive postaccident samples from the RCS, the containment sump, and the containment atmosphere. The applicant stated in DCD Tier 2, Section 1.9.3, that the AP1000 design is consistent with this guidance. Therefore, the staff finds the applicant's elimination of the PASS for the AP1000 design to be acceptable.

As discussed in Section 13.3.3.4 of this report, the PASS requirements of Issue II.B.3 have been eliminated by the model safety evaluation. The staff concludes that, based on the evaluation in Sections 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

Item II.B.8 discusses 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 concludes that vendors should address severe accidents during the design stage to take full advantage of the insights gained from 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 is much more cost effective than modifying existing plants.

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

- NRC Policy Statement, “Severe Reactor Accidents Regarding Future Designs and Existing Plants,” issued August 8, 1985

- NRC Policy Statement, “Safety Goals for the Operations of Nuclear Power Plants,” issued August 4, 1986

- NRC Policy Statement, “Nuclear Power Plant Standardization,” issued September 15, 1987

- 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 Current Regulatory Requirements,” dated January 12, 1990, and the corresponding SRM dated June 26, 1990

- SECY-93-087, “Policy, Technical And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,” dated April 2, 1993, and the corresponding SRM dated July 21, 1993

Generic Issues

The NRC policy statements 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 offer Commission-approved positions for implementing features in new designs for preventing severe accidents and mitigating their effects.

SECY-93-087 and 10 CFR Part 52 serve as the basis for resolving severe accident issues associated with the AP1000. Title 10, Part 52, of the Code of Federal Regulations requires (1) compliance with the TMI requirements in 10 CFR 50.34(f), (2) resolution of USIs and GSIs, and (3) completion of a design-specific PRA. 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 of the staff's deterministic evaluation of severe accident performance for the AP1000 design. The staff evaluates the AP1000 design 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: Testing Requirements

As discussed in NUREG-0933, Issue II.D.1 addresses 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.

The EPRI conducted a safety and relief valve test program for a group of PWR licensees to respond to the staff recommendations in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979, 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 Issue II.D.1 requirements. The results were documented in EPRI-NP-2770-LD, "EPRI PWR Safety Valve Test Report," issued in 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," issued in 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 stated 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 a design similar 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 applicant's commitment in DCD Tier 2, Section 1.9.3, is consistent with the acceptance criteria used by the staff in its evaluations of Issue II.D.1 for operating plants, and is acceptable for the AP1000 design. Therefore, Issue II.D.1 is resolved for the AP1000 design.

Issue II.D.3: Relief and Safety Valve Position Indication

As discussed in NUREG-0933, Issue II.D.3 addresses the requirements in NUREG-0737 for positive indication in the MCR of RCS relief or safety valve position. The acceptance criterion for the resolution of this issue is for the plant design to 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 MCR 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 stated 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 that 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) because positive position indication is provided for the pressurizer safety valves and the RNS 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 reclose and prevent further flow of fluid after normal conditions have been restored. The pressurizer safety valves are of the totally enclosed, pop-type. The RNS relief valve is designed for water relief.

Therefore, the staff considers that the AP1000 design satisfies the requirements of Item II.D.3, and this issue 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 addresses improving the reliability of the AFWS or the EFWS. The issue addresses the following requirements in NUREG-0737:

- a simplified EFW system reliability analysis to determine the potential for system failure under various loss-of-main-feedwater transients
- the acceptance criteria in SRP Section 10.4.9 and BTP Auxiliary Systems Branch (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 stated that the AP1000 design does not utilize an AFWS. A non-safety-related SUFWS removes 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 with mitigating licensing DBAs, as discussed in DCD Tier 2, Chapter 15. The safety-related PXS provides emergency core decay heat removal during transients, accidents, or whenever the normal non-safety-related heat removal paths are unavailable; DCD Tier 2, Section 6.3, describes this system.

On the basis of its review, 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: Auxiliary Feedwater Automatic Initiation and Flow Indication

As discussed in NUREG-0933, Issue II.E.1.2 addresses improving the reliability of the AFWS or EFWS. It discusses 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 AFWS initiation and for flow indication in the MCR.

In DCD Tier 2, Section 1.9.3, Items (1)(ii) and (2)(xii), the applicant stated that the AP1000 design includes the non-safety-related SUFWS and not an AFWS. 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 PXS 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 AFWS. The design employs 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 Std 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” which superseded IEEE Std 279-1971. 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 PXS provides for emergency core cooling during transients and accidents, when the normal heat removal paths are not available.

Although the AP1000 design does not have a safety-related AFWS, it provides the SUFWS, which adequately addresses the requirements in this issue as discussed in Section 10.4.9 of this report.

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

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

As discussed in NUREG-0933, Issue II.E.1.3 addresses improving the reliability of the AFWS or the EFWS. The NRC planned to update SRP Section 10.4.9 and revise RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for Comment)" to include these systems and possibly endorse certain standards. The NRC updated the SRP section in July 1981 but found no additional public and occupational risk reduction to support the need to revise the RG, and thus did not revise it. This issue is resolved because the changes to the SRP established the requirements.

In DCD Tier 2, Section 1.9.4.2.1, the applicant stated that this issue was a requirement to update SRP Section10.4.9 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 I 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 I water source for either the main or startup feedwater systems. The PRHR system provides the safety-related function to remove heat from the primary system when the main feedwater is not available. The bases of the design criteria for the SUFWS are operational and investment protection considerations and not the requirements of SRP Section 10.4.9 or RG 1.26.

The SUFWS does not have to meet the requirements of SRP Section 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 addresses the NRC research programs focused on SBLOCAs and reactor transients. The programs include 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 NRC completed the programs called for in this issue. The programs showed that the ECCS will provide adequate core cooling for SBLOCAs and anomalous transients consistent with the single-failure criteria of 10 CFR Part 50, Appendix K. Issue II.K.3(30) in this section addresses the application of the experimental data from the research programs to validate the conservatism of the licensing codes used in the SBLOCAs .

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

The AP1000 design is a PWR with passive safety systems evolved from the AP600 design, which was the first passive ALWR design reviewed by the NRC. The distinguishing feature of this 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, 10 CFR 52.47(b)(2)(i)(A) imposes the following requirements:

- 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 of 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 nonpassive LWRs applies 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 for their operation. Further, these “passive” systems still rely on some active operation to place them in operation.

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

As described in Chapter 21 of this report, the staff has evaluated and concluded that the applicant’s earlier test program conducted for the 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 provide additional test data on 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 DC. The applicant proceeded to obtain experimental data using the APEX-AP1000 facility which were used to assess models in their thermal-hydraulic codes. These data were used in code assessment, and identified the need to perform a bounding calculation to conservatively account for the effects of liquid entrainment. Chapter 21 of this report discusses the staff evaluation of the APEX-AP1000 test program in support of AP1000. The staff considers Issue II.E.2.2 resolved for the AP1000 design.

Issue II.E.3.1: Reliability of Power Supplies for Natural Circulation

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

The safety-related PXS can establish and maintain natural circulation cooling using the PRHR HX, transferring the decay heat to the IRWST and to the PCS 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 Penetrations

The AP1000 DSER regarding Issue II.E.4.1 stated the following:

> This issue remains open because DCD Tier 2 does not comply with current regulations for the control of combustible gas in containment during accidents.
>
> 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,dated 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.
>
> The applicant wrote DCD Tier 2 in anticipation of these rule changes. Thus, it does not comply 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 Open Item 6.2.5-1.

Subsequent to the publication of the DSER, the NRC revised its regulations regarding the control of combustible gas in containment. The revised regulations were published on September 16, 2003, and became effective on October 16, 2003. The NRC has extensively revised 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” made associated changes to Subsections 50.34 and 52.47, and added a new section, Subsection 50.46a, “Acceptance Criteria for Reactor Coolant System Venting Systems.” The revisions apply to current power reactor licensees, and also consolidate combustible gas control regulations for future power reactor applicants and licensees. The revised rules eliminate the requirements for hydrogen recombiners and hydrogen purge systems, and relax the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance.

TMI Action Plan Requirement II.E.4.1 of NUREG-0737 stated that plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should have containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of GDC 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system. However, as stated above, the revised rules eliminate the requirements for hydrogen recombiners and hydrogen purge systems, and therefore Issue II.E.4.1 has been closed generically.

In conclusion, Open Item 6.2.5-1 is closed and Issue II.E.4.1 is resolved for the AP1000 design.

Issue II.E.4.2: Isolation Dependability

As discussed in NUREG-0933, Issue II.E.4.2 addresses 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 addresses the need for dependable isolation of containment penetrations.

In DCD Tier 2, Section 1.9.3, Item (2)(xiv), the applicant stated 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 I piping systems not connected to the RCS or to the containment atmosphere. The design incorporates fewer containment penetrations than 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 acceptance criteria for SRP Section 6.2.4 encompass the Issue II.E.4.2 requirements. The staff 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. In addition, Issue II.E.4.4 restricts 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, Section 1.9.3, states that the AP1000 will meet the Issue II.E.4.4 requirement. DCD Tier 2, Section 6.2.3.1.3.F, states, “Isolation valves are designed to have the capacity to close against the conditions that may exist during events requiring containment isolation.” Technical specifications will preclude unnecessary venting. The AP1000 design provides debris screens 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 addresses 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. As stated in 10 CFR 50.34(f)(2)(xvi), the applicant should establish a design criterion 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.

DCD Tier 2, Section 1.9.3, states that this issue applies only to B&W designs. The AP1000 design uses the PXS 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 PXS. 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, includes additional information.

In the staff's 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: Test Adequacy Study

As discussed in NUREG-0933, Issue II.E.6.1 addresses 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. The staff divided this issue into the following four parts during its resolution:

- testing of pressure isolation valves (PIVs)
- in situ testing and surveillance of check valves
- reevaluation of the thermal-overload protection provisions for MOVs in RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"
- operability verification for MOVs in accordance with GL 89-10

Section 3.9.6.2 of this report discusses the staff's evaluations of the first two parts, testing of PIVs and check valves. The staff also discussed the last two parts in the list above in Section 3.9.6.2 of this report as a part of the resolution of GL 89-10. Since the resolution of the four separate parts listed above subsumed the resolution of Issue II.E.6.1, and on the basis of the staff's evaluations and resolution of the issues 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 addresses the provision of 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 MCR 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 issued requirements. The radiation and noble gas effluent instrumentation must 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, Revision 3, dated May 1983. NUREG-0660 also provides the requirements for a human factors analysis, which includes the operator’s use of the indicators listed above 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.

In DCD Tier 2, Section 1.9.3, Item (2)(xvii), the applicant stated that DCD Tier 2, Chapter 7 describes the AP1000 postaccident monitoring system.

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 compared the AP1000 design to the criteria in RG 1.97 and addressed accident monitoring instrumentation.

The hydrogen monitors are not part of postaccident monitoring. Other noble gas effluents are designated Type E variables and include information that permits the operator to take the following actions:

- monitor the habitability of the MCR

- 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, Section 11.5.5, includes additional discussion on the measurement of radioactive effluents and conformance of the AP1000 design 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 Gray (Gy) per hour (/hr) (10^7 Rad per hour (R/hr) in the containment following an accident. In DCD Tier 2, Section 11.5.6.2, the applicant stated that the AP1000 design will incorporate four electrically independent ion chambers located inside the containment to measure high-range gamma radiation. These detectors, mounted on the inner containment wall in widely separated locations, will have an unobstructed "view" of a representative volume of the containment atmosphere. The design and qualification of these monitors comply 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 (as addressed in DCD Tier 2, Section 9.3.3.1). Chapter 18 of this report addresses the human factors aspects of this issue. Section 7.5 of this report discusses accident monitoring instrumentation.

Therefore, Issue II.F.1 is resolved for the AP1000 design.

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

10 CFR 50.34(f)(2)(xviii) requires that instruments be provided in the MCR. These instruments must 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. The TMI Action Plan Item II.F.2 discusses the ICC phenomena and the need to have a reactor water level indication system that displays the 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 at 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 PORVs stuck in the open position, 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, the continued operation of the RCPs maintained core cooling. 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 lessons learned from TMI, the reactor vessel water level indication system was added, specifically for PWRs, to ensure operator action to trip the RCPs following a LOCA earlier 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 that addresses ICC.

In DCD Tier 2, Section 1.9.3(2)(xviii), the applicant stated 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 listed the AP1000 features that provide margin to or indication of ICC, with additional information provided in DCD Tier 2, Sections 6.3 and 7.5.

In response to 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 the CMT, ADS, PRHR, and IRWST to maintain core cooling in the event of a SBLOCA. The AP1000 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. In the AP1000, the operator does not need to (1) trip the RCPs, (2) inject water into the core, or (3) manually depressurize the plant during an SBLOCA.

The instruments typically used in current PWRs include subcooling margin monitoring capability, core exit thermocouples, and a reactor vessel level indication system, which together give the operator the ability to monitor the coolant conditions and to act appropriately to ensure core cooling during the approach to the ICC conditions and to recover from those 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’s 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 can indicate 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 reliably indicate ICC. The AP1000 hot-leg water level indication does not 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. However, the hot-leg level indication system does verify reactor water inventory to terminate the recovery action in the ERGs for the ICC event.

Because the AP1000 design automatically trips the RCPs during an SBLOCA event, and because the operators are unlikely to be misled by forced two-phase flow, the core exit temperature is an important and sufficient indication of an approach to ICC conditions. The temperature reading provided by core exit thermocouples is appropriately included in the ERGs for plant recovery.

The staff reviewed the applicant’s response and determined that for an 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 unambiguously indicate an approach to ICC, and the safety-related hot-leg level indication is used only to terminate the recovery action in the ERGs for the ICC event.

Therefore, the applicant has satisfied the requirements for ICC, as discussed in 10 CFR 50.34(f)(2)(xviii). 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, the accident monitoring instrumentation in nuclear power generating stations followed the guidance 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 instrumentation will be of sufficient quantity, range, availability, and reliability to permit adequate monitoring of plant variables and systems during and after an accident. Specifically, the instrumentation should 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, ESF 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 stated that the AP1000 postaccident monitoring system is designed using RG 1.97 as a guidance document. Either the normal MCR display system or the QDPS displays data used for postaccident monitoring. The normal MCR display system displays non-safety-related signals which need not be displayed by a qualified system. The QDPS provides for the display of signals that must be displayed by a qualified system. The QDPS 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. The preceeding response for Issue II.F.1 and DCD Tier 2, Section 7.5, provides additional information.

On the basis of the above and the staff’s review in Section 7.5.9 of this report, 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 addresses upgrading the emergency power for the pressurizer relief and block valves and pressurizer level indicators. In accordance with the requirements in NUREG-0737, an emergency source of power must supply the pressurizer equipment in the event of a LOOP.

In DCD Tier 2, Section 1.9.3, the applicant stated 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 above, 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 addresses 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. Issue I.B.1.1, "Organization and Management Long Term Improvements," which was resolved with changes to RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," and RG 1.33, "Quality Assurance Program Requirements (Operation)," included this issue.

In DCD Tier 2, Section 1.9.3, Item (3)(vii), the applicant stated that it has devised a management plan 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 COL applicant will determine the procedures to be used in the construction, startup, and operation phases of the AP1000.

The organization of the plant beyond the AP1000 design, the construction of the plant, and the modification of the plant is outside the scope of DC for the AP1000 design. In part, these concerns involve the organization of the COL applicant; however, the concerns about 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.

Chapter 17 of this report discusses and finds acceptable the QA standards and the organization that the applicant used for the design of the AP1000. Furthermore, in DCD Tier 2, Table 1.9-2, the applicant stated 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 addresses assuring that the applicant promptly report all reportable items and submit complete information to the NRC. The issue was resolved when the NRC issued new requirements in 10 CFR Part 21, "Reporting of Defects and Noncompliance," and 10 CFR 50.55(e), on July 31, 1991.

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

As indicated in Section 18.3, 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) requests 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 ERG AES-0.2, “Natural Circulation Cooldown.” There are ERGs that direct the operators to cool down 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 to direct the operators to verify that a steam void does not exit in the vessel. On the basis of this review, the staff concludes that the ERGs direct 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) asks licensees to provide operating procedures to ensure that operators will not rely on level indication alone in evaluating plant conditions. As stated in NUREG-0933, the staff determined that Issues I.A.3.1, I.C.1, and II.F.2 cover this issue and that this issue is resolved.

The NRC implemented Issue I.A.3.1, “Revise Scope and Criteria for Licensing Examinations,” 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 the COL applicant referencing the AP1000 design; therefore, the applicant does not have to address Issue I.A.3.1 for compliance with 10 CFR 52.47(a)(1)(iv).

DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue is not relevant to the AP1000 design because it is not a DC issue, but is the responsibility of the COL applicant. This is COL Action Item 20.1.4-1. However, the applicant has earlier stated that the proposed resolution to Issues I.C.1 and II.F.2 addresses the design portion of this item.

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. Operators determine the status of core cooling by indications of core-exit thermocouple temperature, RCS subcooling, and RCS hot-leg 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) addresses 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 AFW valves are in the open position. This issue was resolved and NUREG-0737 issued additional requirements.

DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue is not relevant to the AP1000 design because it is the responsibility of the COL applicant. This is COL Action Item 20.1.4-1.

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

<u>Issue II.K.1(10): Review and Modify Procedures for Removing Safety-Related Systems from Service</u>

As discussed in NUREG-0933, Issue II.K.1(10) addresses 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.

DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue was not relevant to the AP1000 design because it is the responsibility of the COL applicant. This is COL Action Item 13.5-1.

As indicated in Section 18.3, of this report, the applicant satisfactorily addressed this item in WCAP-14645. The staff concludes that WCAP-14645 is applicable to the AP1000 design; therefore, Issue II.K.1(10) is resolved.

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

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

In DCD Tier 2, Section 1.9.4.2.1, Westinghouse stated that it based the AP1000 TS (DCD Tier 2, Section 16.1) on and reviewed them against the Westinghouse standard TS (STS), which incorporates the requirements of the BLs for the TMI Action Plan. Chapter 16 of this report evaluates the AP1000 TS. The staff reviewed the AP1000 TS against the STS. On the basis of this review, the staff concludes that the approved AP1000 TS incorporate all the appropriate BL requirements from the TMI Action Plan. Section 20.7 of this report discusses the incorporation of operating experience in BLs in the AP1000 design.

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

<u>Issue II.K.1(16): Implement Procedures That Identify Pressurizer (PZR) PORV "Open" Indications and That Direct Operators to Close Valve Manually at "Reset" Setpoint</u>

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

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

<u>Issue II.K.1(17): Trip Pressurizer Level Bistable So That Pressurizer Low Pressure Will Initiate Safety Injection</u>

As discussed in NUREG-0933, Item II.K.1(17) addresses the requirement for the applicant's plants to trip the PZR level bistable so that the PZR low pressure, rather than the PZR low pressure and pressurizer low-level coincidence, will initiate safety injection.

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

<u>Issue II.K.1(22): Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Is Not Operable</u>

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

The applicant stated in DCD Tier 2, Table 1.9-2, that it considers Issue II.K.1(22) relevant to the AP1000 design; however, resolution of this issue is not necessary 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 stated that, although this issue applies only to BWRs in NUREG-0660, there are some considerations for the AP1000 design. Following a loss of main feedwater (LMFW), a number of plant systems automatically actuate to provide decay heat removal. The non-safety-related DGs can power the non-safety-related SUFWS, which is automatically actuated and controlled by low SG level. For design-basis events, the safety-related PXS includes PRHR HXs, 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 an LMFW event. In DCD Tier 2, Sections 6.3 and 10.4 provide additional information.

On the basis of its review, which is discussed in Section 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 an LMFW, and the PXS 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 requires 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 Issue I.C.1 covered this issue for PWRs.

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 they have 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 considers 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 Issue I.C.1 covers this issue.

DCD Tier 2 does not address this issue. The applicant indicated, in DCD Tier 2, Table 1.9-2, that one or more issues have superseded this issue. As stated above, Issue I.C.1 covers this issue, and the applicant considers Issue I.C.1 to be 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 in Section 20.4 of this report, the staff has completed its review of Issue I.C.1 and concludes that Issue I.C.1 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 Issues I.A.3.1, “Revise Scope of Criteria for Licensing Examinations,” I.C.1, and I.G.1, cover this issue.

As stated in NUREG-0933, the staff has implemented Issues I.A.3.1, I.C.1, and I.G.1 in its review of reactor plant designs, and the applicant does not need to address them for compliance with 10 CFR 52.47(a)(1)(iv).

DCD Tier 2 does not address this issue. The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue is not relevant to the AP1000 design because one or more other issues have superseded it. As stated above, Issues I.A.3.1, I.C.1, and I.G.1 cover this issue, and the applicant considers these issues to be the sole responsibility of the COL applicant.

An earlier part of this section addressed Issue 1.C.1. Issue I.A.3.1 involved the revision of the scope of examinations and criteria for licensing examinations, and Issue I.G.1 concerned new training requirements for operators. In DCD Tier 2, Table 1.9-2, the applicant identified Issues I.A.3.1 and I.G.1 as the responsibility of the COL applicant rather than the responsibility of the applicant in the AP1000 design review.

The preceding section discusses the guidelines developed as part of Issue II.K.1(25). Also, as indicated in Section 18.3, of this report, the applicant satisfactorily addressed Issue I.C.1 of this item in WCAP-14645, because the staff has concluded that WCAP-14645 is applicable to the AP1000 design.

However, the staff also identified COL Action Item 20.4-5 for Issue II.K.1(26). This action item requires that the COL applicant address the scope of 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.

<u>Issue II.K.1(27): Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions</u>

As discussed in NUREG-0933, Issue II.K.1(27) addresses the need for PWR licensees to provide analyses and to develop guidelines and procedures for an ICC condition. The staff determined in NUREG-0933 that Issues I.C.1 and II.F.2 cover this issue. An earlier part of this section discussed the resolution of Issues I.C.1 and II.F.2 for the AP1000 design.

DCD Tier 2 does not address Issue II.K.1(27). The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue is not relevant to the AP1000 design because one or more other issues have superseded it. Although, as stated above, Issues I.C.1 and II.F.2 cover this issue, the applicant considers it the sole responsibility of the COL applicant. The applicant addressed Issue II.F.2 in DCD Tier 2, Section 1.9.3, Item (2)(xviii).

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

The staff concludes that the applicant has provided appropriate 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) addresses the requirement that PWRs be designed to ensure automatic RCP trip for all circumstances where required. The staff determined in NUREG-0933 that Issue II.K.3(5), "Automatic Trip of Reactor Coolant Pumps," covers this issue.

DCD Tier 2 does not address Issue II.K.1(28). The applicant concluded, in DCD Tier 2, Table 1.9-2, that this issue is not relevant to the AP1000 design because one or more other issues (i.e., Issue II.K.3(5)) had superseded it. In DCD Tier 2, Section 1.9.4.2.1, the applicant responded to Issue II.K.3(5), stating that the AP1000 design provides for an automatic trip of the RCPs on actuation of the PXS. This trip is provided to prevent RCP interaction with the operation of the CMT. The staff concludes that Issue II.K.1(28), as well as II.K.3(5), 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) addresses the requirement for B&W plants to provide a design and schedule for implementation of a safety-grade reactor trip on LMFW, turbine trip, and significant reduction in SG level. These requirements appear as Item 5 in NRC bulletin, BL 79-05B, "Nuclear Incident at Three Mile Island - Supplement," issued on April 21, 1979. The NRC resolved this issue and established new requirements.

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

The applicant further stated 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 applicant's 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) requires licensees to evaluate the likelihood and consequences of RCP seal damage following an SBLOCA with a loss of LOOP.

In DCD Tier 2, Section 1.9.3, Item (1)(iii), the applicant stated 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 Issue 23, covers this issue. 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) addresses the requirement in NUREG-0737 and NUREG-0660 for PWR operating plants to provide a system that uses the PORV block valves to protect against SBLOCAs. 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 the establishment of new regulatory requirements and/or guidance. In DCD Tier 2, Section 1.9.3, Item (1)(iv), the applicant stated that the AP1000 design does not include PORVs. The PZR volume is about 40 percent larger than the PZR 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 PZR surge line is also larger to permit a more rapid transfer of coolant between the RCS and the PZR and to accommodate the ADS 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.

The applicant also stated that two totally enclosed pop-type safety valves provide overpressure protection. These valves are spring-loaded and self-actuated, and they are designed to meet the requirements of ASME Code, Section III. If the PZR 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 when the valves open. The AP1000 design includes an ADS consisting of six parallel sets of two valves in series connected to the PZR 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 because 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) requires applicants to document the action to be taken to decrease the probability of an SBLOCA caused by a stuck-open PORV. The design purpose of PORVs is to prevent RCS overpressure and to reduce the 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 the establishment of new regulatory requirements and/or guidance. In DCD Tier 2, Section 1.9.3, Item (1)(iv), the applicant stated that the AP1000 design does not include PORVs. It further described the AP1000 PZR design, including the PZR safety valves and

ADS, to justify the appropriateness of the AP1000 design. (See the discussion in Issue II.K.3(1) above.)

On the basis of the preceding discussion and because 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 Reactor Coolant Pumps

As discussed in NUREG-0933, Issue II.K.3(5) addresses 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. Licensees should consider the preservation of the maximum RCS inventory in the SBLOCA mitigation and the most effective DHR strategy for the mitigation of other transients.

In DCD Section 1.9.4.2.1, the applicant stated that the AP1000 design provides for an automatic trip of the RCPs upon actuation of the PXS. This trip is provided to prevent RCP interaction with the operation of the CMT. DCD Tier 2, Section 6.3, provides additional information regarding the automatic RCP trip.

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) addresses the requirement for licensees to provide instrumentation to verify natural circulation during transient conditions. The staff determined in NUREG-0933 that Issues I.C.1, II.F.2, and II.F.3 cover this issue.

The applicant does not address this issue in Section 1.9.4.2.1 because DCD Tier 2, Table 1.9-2, indicates that other issues supersede it.

On the basis of its review of the compliance of the AP1000 design with Issues I.C.1, II.F.2, and II.F.3, as described in this report, the staff concludes that those issues relevant to the resolution of the TMI Action Item II.K.3(6) have been resolved. The respective TMI item discussions provide more detailed information. 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(8) addresses further staff consideration of the need for diverse decay heat removal methods, which are independent of the SGs. The staff has determined in NUREG-0933 that Issues II.C.1, "Interim Reliability Evaluation Program," and II.E.3.3, "Coordinated Study of Shutdown Heat Removal Requirements," cover this issue. In NUREG-0933, the staff also stated that Issue A-45, "Shutdown Decay Heat Removal Requirements," addressed Issue II.E.3.3.

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 other issues supersede it. As stated in NUREG-0933, the staff implemented

Issues A-45, II.C.1, and II.E.3.3 in its review of reactor plant designs, and the applicant and the staff do not have to address these issues to demonstrate compliance with 10 CFR 52.47(a)(1)(iv).

In DCD Tier 2, Appendix 19E, the applicant described the AP1000 shutdown evaluation. Appendix 19E describes multiple decay heat removal capabilities independent of the SG. Chapter 19.3 of this report includes detailed discussion of the multiple decay heat capabilities. On the basis of 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) addresses requiring the applicant's plants to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs. NUREG-0737 and NUREG-0660 issued the requirements.

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

The applicant addressed this issue in DCD Tier 2, Section 1.9.4.2.1, Item II.K.3(9), where it stated that this issue is not applicable to the AP1000 design because the design does not have PORVs. DCD Tier 2, Sections 5.1.2 and 5.2.2, provide additional information.

On the basis of 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) addresses requiring BWR plants to modify the ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. NUREG-0737 and NUREG-0660 issued the requirements for a feasibility study and risk assessment study to determine the optimum approach.

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 the establishment of new regulatory requirements and/or guidance resolves this issue. In DCD Tier 2, Section 1.9.3, Item (1)(vii), the applicant stated 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. The applicant stated that manual actuation of the ADS is not required to maintain core cooling. As discussed under Issue II.B.8 in this section 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. The staff evaluates this reliability in Section 19.1 of this report.

The actuation of ADS Stages 2 and 3 occurs on a set time delay after the actuation of the first stage, as discussed above. Stage 4 of the ADS 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's 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 Seals

As discussed in NUREG-0933, Issue II.K.3(25) requires 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 LOOP should prevent excessive loss of RCS inventory following an anticipated operational occurrence. NUREG-0737 requires 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 in DCD Tier 2, Table 1.9-2, that the establishment of new regulatory requirements and/or guidance resolves this issue. In DCD Tier 2, Section 1.9.3, Item (1)(iii), the applicant stated 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 Issue 23, which is discussed in Section 20.3 of this report, covers this issue. 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) addresses 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. NUREG-0737 and NUREG-0660 issued the requirements.

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 the establishment of new regulatory requirements and/or guidance resolves this issue. In DCD Tier 2, Section 1.9.3, Item (1)(x), the applicant stated that although this issue is identified as applicable to BWRs only, the AP1000 uses a safety-related ADS that differs 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. The operation of these valves does not rely on non-safety-related equipment or instrumentation, including instrument air or nitrogen supply. These valves are designed and qualified to function in the conditions of an accident, they will be subject to preoperational and inservice testing, and they will be part of the RAP. On the basis of the staff's review, Issue II.K.3(28) is resolved for the AP1000 design.

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

As discussed in NUREG-0933, Issue II.K.3(30) requires licensees to revise and submit analytical methods for SBLOCA analyses for compliance with 10 CFR Part 50, Appendix K, for NRC review and approval. The revised LOCA methods were to account for comparisons with experimental data, including data from LOFT and Semiscale test facilities. Alternatively, licensees were to provide additional justification for the acceptability of their SBLOCA models with LOFT and Semiscale test data. NUREG-0737 contained clarifications.

DCD Tier 2, Table 1.9-2, indicates that this issue is not an AP1000 DC issue because the issue applies to currently operating plants or is the responsibility of the COL applicant. This is COL Action Item 20.1.4-1.

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

Issue III.A.1.2: Upgrade Licensee Emergency Support Facilities

As discussed in NUREG-0933, Issue III.A.1.2 addresses requiring licensees to upgrade their emergency support facilities by establishing a TSC, an operational support center (OSC), and a near-site emergency operations facility (EOF) for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. NUREG-0737, and "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability, (GL 82-33)," dated December 17, 1982, resolved this issue and issued new requirements.

The applicant indicated, 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 stated 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. In DCD Tier 2, Sections 18.8.3.5 and 18.8.3.6 described the mission, major tasks, and location of the TSC and OSC for the AP1000 standard design. Figures 1.2-19 and 1.2-18 show the location of the TSC and OSC, respectively. The applicant stated 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, the standard design must consider certain features, facilities, functions, and equipment necessary for emergency planning. Specifically, in accordance with 10 CFR 50.34(f)(2)(xxv), the standard design must address the characteristics of the onsite TSC and OSC. The staff's review documented two open items (13.3-1 and 13.3-2) associated with the AP1000 emergency support facilities in the DSER. These open items are now resolved. Therefore, the staff concludes that, based on the evaluation in Section 13.3.3 of this report, and specifically the resolution of the open items noted above, Issue III.A.1.2 is 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 addresses the need for licensees to upgrade their communications capability at the emergency support facilities at the plant listed in Issue III.A.1.2. NUREG-0660 contains relevant guidance.

Generic Issues

DCD Tier 2, Section 13.3, states that emergency planning, including communication interfaces among the MCR, the TSC, and the emergency planning centers, is 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 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. The COL applicant will also address the emergency response facility communication system, including the crisis management radio system.

The staff considers this issue to be outside the scope of the AP1000 DC and, therefore, the COL applicant will address it. DCD Tier 2, Section 13.3, addresses this issue and Section 13.3.2 of this report discusses it 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 addresses 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:

(1) III.D.1.1(1), "Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Plants"

(2) III.D.1.1(2), "Review Information on Provisions for Leak Detection"

(3) 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. This section addresses Issue II.F.1 for the AP1000 design.

The staff reviewed the need for requiring leak detection systems and the development of new acceptance criteria for those systems encompassed by Subissue III.D.1.1(3) as part of other issues, as described in Subissue III.D.1.1(2). Therefore, work on Subissue III.D.1.1(3) did not provide any data for staff consideration, and the staff dropped this issue from further consideration.

In DCD Tier 2, Section 1.9.3, Item (2)(xxvi), the applicant stated 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 radiation levels are high within the containment. On the basis of the staff's review, Issue III.D.1.1 is resolved for the AP1000 design.

Issue III.D.3.3: In-Plant Radiation Monitoring

As stated in 10 CFR 50.34(f)(2)(xxvii) (III.D.3.3), the licensee shall "provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

Area and airborne radiation monitors at the AP1000 will 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, "Guide for Administrative Practices in Radiation Monitoring," 8.8, "Information Relevant to Ensuring that Occupation Radiation Exposures at Nuclear Power Station Will Be As Low As Is Reasonably Achievable."

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 stated 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. This is COL Action Item 12.4.4-1.

The information on in-plant radiation monitoring in DCD Tier 2, Chapter 12, 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 addresses upgrading the habitability of the control room for the operators. NUREG-0737 provided the requirements.

In DCD Tier 2, Section 1.9.3, Item (2)(xxviii), the applicant stated 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 applicant stated that MCR is 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.

The applicant also stated that the non-safety-related HVAC system, equipped with a refrigeration-type air conditioning unit and powered from the onsite DGs, 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 a letter dated May 21, 2003, the applicant committed to conform to the guidance of RG 1.78, Revision 1, to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19. In addition, the applicant agreed to revise the DCD to refer to RG 1.78, Revision 1. Confirmatory Item 6.4-1 in the DSER identified the need for inclusion of this information in the DCD. The staff has reviewed the DCD and concludes that it appropriately refers to RG 1.78, Revision 1. Therefore, Confirmatory Item 6.4-1 is resolved.

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 I, Class 1E toxic gas monitoring, as required. It also states that RG 1.78, Revision 1, addresses control room protection for toxic chemicals and evaluation of offsite toxic releases (including the potential for toxic releases beyond 72 hours) in order to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19. This is COL Action 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. DCD Tier 2, Section 15.6.5.3, detailed the analysis assumptions for modeling the doses to MCR personnel. Open Item 6.4-1 in the DSER, identified that the staff could not complete its review and independent dose assessment until it had resolved questions on the assumed aerosol removal rates in the containment.

In Section 6.4 of this report, the staff resolved Open Item 6.4-1 and found that the VES, under "high-high" radiation conditions as described in the AP1000 DCD Section 6.4, is capable of mitigating the dose in MCR following DBAs to meet the dose criteria specified in GDC 19 as applied to the AP1000 design.

Therefore, Issue III.D.3.4 is resolved for the AP1000 design.

20.5 Human Factors Issues

Issue HF1.1: Shift Staffing

This issue addresses 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, an applicant must consider 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; and appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation.

The review criteria for this issue appear in 10 CFR 50.54, SRP Sections 13.1.2–13.1.3, “Operating Organization,” 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 DCD Tier 2. It concludes, in DCD Tier 2, Table 1.9-2, that this issue is not relevant to the AP1000 design because it is the responsibility of the COL applicant. This is COL Action Item 20.1.4-1. As indicated earlier in this report in Section 18.3, 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 addresses the development of criteria by the NRC to provide assurance during inspections that operating plant EOPs are adequate and can be used effectively. The staff published lessons learned from its inspections of EOPs at plants in NUREG-1358, “Lessons Learned from the Special Inspection Program for Emergency Operating Procedures,” April 1989. The NRC later issued Temporary Instruction (TI) 2515/92, “Emergency Operating Procedures Team Inspections,” containing guidance for conducting these inspections.

The issue was resolved with no new requirements. In DCD Tier 2, Section 1.9.4, the applicant stated 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, has additional information. DCD Tier 2, Section 18.9, “Procedure Development,” covers this issue, which is a COL responsibility. This is COL Action Item 13.5-1. 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 effective, 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 and abnormal operating procedures. To satisfy the objective of this issue, an applicant must (1) develop guidelines for preparing and criteria for evaluating normal operating procedures and other procedures that affect plant safety and (2) upgrade the procedures, train the operators in their use, and implement the upgraded procedures.

The review criteria for this issue appear in SRP Sections 13.5.1, “Administration Procedures,” and 13.5.2, “Operating and Maintenance Procedures,” and in Information Notice (IN) 86-64, “Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures.” In addition, Element 8, “Procedures Development,” of NUREG-0711, “Human Factors Engineering Program Review Model” covers this item.

As indicated in Section 18.3 of this report, the applicant satisfactorily addressed this item in WCAP-14645, and the staff has concluded that WCAP-14645 is applicable to AP1000 design. On the basis of the staff’s review, Issue HF4.4 is resolved for the AP1000 design.

Generic Issues

Issue HF5.1: Local Control Stations

As discussed in NUREG-0933, Issue HF5.1 addresses 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 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 1.9.4.2.4, the applicant stated that it has used techniques and experience gained in the design of the MCR and remote shutdown panel on the local control station panels. The methodology for analyzing the job/tasks of the control room is applied to the analysis of 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 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 Advanced Controls and Instrumentation

As discussed in NUREG-0933, Issue HF5.2 addresses 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. The NRC did not believe these guidelines would be adequate for advanced and developing technologies that could be introduced into future designs. The agency expected that improved alarm systems using advanced technologies would become available, and regulators would develop guidelines for the use and evaluation of these longer term alarm improvements.

This issue focused on the potential risk that could result from the human error in the use of control room alarms. The staff stopped work on this issue when the Office of Nuclear Regulatory Research (RES) integrated the development of review guidance for advanced alarms into its program to develop an “Advanced Human-Interface Design Review Guideline.”

This issue is resolved with no new requirements. In DCD Tier 2, Section 1.9.4.2.4, the applicant stated that the AP1000 advanced alarm design, described in DCD Tier 2, Section 18.9.2, conforms with current guidance and requirements for integrated human factors design. WCAP-14690 describes the computerized procedures, and DCD Tier 2, Section 18.8.3.2, gives a detailed description of the QDPS. DCD Tier 2, Section 18.8.2.3, contains the plan for the V&V of the AP1000 man-machine interface system (M-MIS).

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 applicant also explained the relationship of the computerized procedures and QDPS. As indicated in Section 18.3 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 DC must demonstrate compliance with any technically relevant TMI requirements in 10 CFR 50.34(f). Table 20.6-1 of this report lists the relevant TMI Action Plan items, the 10 CFR 50.34(f) requirements, and the section in which they are addressed.

Table 20.6-1 10 CFR 52.47(a)(1)(ii) TMI Action Plan Items

TMI REQUIREMENT	10 CFR 50.34(f)	FSER 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
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

TMI REQUIREMENT	10 CFR 50.34(f)	FSER Chapter/Section
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.1.22	(2)(xxi)	10, 20.4
II.K.2	(2)(xxiii)	20.4
II.K.2(16)*	(1)(iii)	20.4
II.K.3.1	(1)(iv)	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
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 (BLs, GLs, and INs) when it believes a significant safety-related event or condition at one or more facilities potentially applies to other facilities. The NRC staff typically issues a BL or GL when it determines that licensees should be required to inform the NRC what actions they have taken or will take to address an event, condition, or circumstance that is both potentially significant to safety and generic. The staff typically issues an IN when it determines that licensees should be informed of an event, condition, or circumstance that is both potentially significant to safety and generic, but the event, condition, or circumstance is not sufficiently significant to warrant requiring licensees to confirm their actions in writing. Potential safety issues highlighted in NRC generic communications have resulted in the establishment of a USI or a 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 DC discuss how it has incorporated operating experience into the design.

The staff examined the AP1000 design for incorporation of important lessons learned from operating plant experience by reviewing the BLs and GLs issued between January 1, 1980, and January 31, 2002, and determining whether the applicant had properly incorporated into the AP1000 design the staff positions in those documents that were applicable. In the NRC programs that account for operating experience, the BLs 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 (LERs)), issue staff positions on resolving problems in these malfunctions, and request licensee actions. In contrast, INs do not request actions on the part of the licensees. Thus, reviewing how the applicant has incorporated pertinent BLs and GLs into the AP1000 design is a sufficient basis for reviewing the design against operating experience.

In the resolution of BLs and GLs for the AP1000 design, the staff looked beyond the specific purpose of the documents to determine their resolution for the AP1000 design. DCD Tier 2, Section 1.9.5.5 states that the applicability of each GL and BL to the AP1000 is assessed in WCAP-15800. WCAP-15800 states that BLs and GLs identified as procurement, procedural, or maintenance/surveillance Issues will be reviewed and addressed by the COL applicant. Items identified as part of COL are also part of this action item. This is COL Action Item 20.7.1-1.

20.7.2 Application Content Review

The applicant submitted DCD Tier 2 for standard plant DC of the AP1000 design. In that document, the applicant stated that the design engineers continually review industry experience from sources such as NRC BLs, LERs, NRC requests for information, and GLs. The applicant also stated that it has incorporated operating plant experience in the AP1000 design by virtue of its participation in developing Volume III of the EPRI ALWR URD and in the activities of the ALWR Utility Steering Committee.

The applicant also submitted WCAP-15800 to address the manner in which it incorporated operating plant experience into the AP1000 design and stated that it had reviewed the NRC BLs, GLs, circulars, INs, and AEOD reports for the period January 1, 1980, to January 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 that the AP1000 design does not have the equipment discussed in the NRC document. The individual documents were separated 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 DC

Generic Issues

The staff considered the applicant's list of BLs and GLs applicable to the AP1000 design in determining its own list of documents to review in examining how the applicant incorporated operating experience in the design.

20.7.2.1 Regulatory Review

The SRP (NUREG-0800) guides the NRC staff in its review of a reactor facility design. This document states requirements, acceptance criteria (some of which the NRC predicates on operating reactor experience), and findings that the staff must make. The NRC last revised this document in April 1982, when it incorporated significant issues raised before January 1981. 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 applicability to the AP1000 design of the BLs and GLs issued by the NRC between January 1, 1980, and December 31, 2002.

As discussed in Section 20.7.1 above, the BLs and GLs address issues of sufficient safety significance to warrant requiring licensees to inform the NRC of the actions they have taken or will take, whereas INs do not require a response. Accordingly, the NRC staff reviewed the BLs and GLs issued between January 1, 1980, and December 31, 2002, applicable to the AP1000 design.

Upon initial review, the NRC excluded certain BLs and GLs from further examination 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.

Sections 20.2 through 20.4 of this report discuss the resolution of the technically relevant generic issues in NUREG-0933 (i.e., TMI Action Plan items, USIs, and GSIs) for the AP1000 design. Tables 20.7-1 and 20.7-2 of this report summarize the resolution of the issues in BLs and GLs, respectively.

20.7.4 Conclusions

The staff selected certain BLs and GLs issued by the NRC between January 1, 1980, and January 31, 2002, for use in its review of the AP1000 design because the issues involved in these documents were not already required by rule, regulation, or policy statement. By NRC letter dated September 2, 1995, the staff listed generic issues (i.e., BLs and GLs) no longer relevant to the AP600 design. The review of these issues duplicate other staff reviews and, therefore were unnecessary for the FSER review of the AP600 design. The staff has reviewed these issues and considers them unnecessary for the FSER review of the AP1000 design. Therefore, the following will not be included in the tables:

BL 80-02, BL 80-03, BL 80-16, BL 86-02, BL 88-03.

GL 80-13, GL 80-16, GL 80-30, GL 80-45, GL 80-48, GL 80-53, GL 80-56, GL 80-82, GL 80-98, GL 80-99, GL 80-100, GL 80-102, GL 80-106, GL 82-08, GL 82-09, GL 82-17, GL 82-23, GL 83-07, GL 83-13, GL 83-26, GL 83-27, GL 83-28, GL 83-30, GL 84-01, GL 84-13, GL 84-24, GL 85-19, GL 86-13, GL 86-15, GL 87-09, GL 88-07,

GL 88-12, GL 88-16, GL 88-18, GL 89-01, GL 89-14, GL 90-02, GL 90-09, GL 91-01, GL 91-04, GL 91-08, GL 91-09, GL 92-08, GL 93-05, GL 93-07, and GL 93-08.

Tables 20.7-1 and 20.7-2 of this report list these BLs and GLs.

On the basis of its review of the BLs and GLs issued between January 1, 1980, and January 31, 2002, and the applicant's report (WCAP-15800) on how these BLs 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	The NRC issued this BL only to BWR licensees to determine the operability of the pneumatic operator for the ADS; however, the AP1000 design has an ADS similar to that in BWRs. In WCAP-15800, Revision 3, the applicant stated that the AP1000 design uses a safety-related ADS that differs from that presently used on BWRs. The AP1000 ADS uses safety-related dc MOVs and squib valves to initiate depressurization. The motive power for these valves is safety-related dc power. The operation of these valves does not rely on any non-safety-related equipment or instrumentation, including instrument air or nitrogen supply. These valves are designed and qualified to function in the conditions of an accident. They will also be the subject of preoperational and inservice testing, and they will be part of the RAP. Based on the foregoing, this BL 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 BL in its review of DCD Tier 2, Sections 15.1.5 and 6.2.1.4, on mass and energy release analysis for a postulated pipe rupture inside containment. This BL asks addressees to review their containment pressure and temperature response analysis to determine if the MSLB accident inside containment included the impact of runout flow from the AFW and the impact of other energy sources, such as continuation of feedwater or condensate flow. It also asks 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 indicated that the effects of startup feedwater flow are maximized in the MSLB 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 have 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 BL 80-04. Therefore, the containment-related aspects of BL 80-04 are 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 MSLB, is addressed in Section 15.2.1.5 of this report. The reactivity-related aspects of this BL 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's review, this BL 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	This BL 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, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," adequately addresses the concern identified in BL 80-05. The bases for the finding are that (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 draindown. DCD Tier 2, Table 11.2-2, shows that the external design pressure for the RCDT of the AP1000 design is 15 psig. The staff confirmed in Section 11.2 of this report that the above bases for the finding in NUREG-1512 are applicable to the AP1000. Therefore, the staff finds that the design of the WLS of the AP1000 as discussed adequately addresses the concern identified in BL 80-05 and, therefore, is acceptable. Based on the staff's review, this BL is resolved for the AP1000.
BL 80-06, Engineered Safety Feature Reset Controls	Westinghouse stated in WCAP-15800, Revision 3, that it addressed this BL in DCD Tier 2, Sections 7.3.1.1, 13.5, and Chapter 14. This BL lists two actions that apply to the AP1000 design—(1) review the I&C system schematics to verify that the ESF equipment remains in its emergency mode upon reset of the ESF actuation signal, and (2) verify that 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 preoperational tests. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 80-08, Examination of Containment Liner Penetration Welds	Westinghouse stated in WCAP-15800, Revision 3, that the bulletin is not applicable to the AP1000 design because the design has no containment liner. The staff agrees with the assessment. Based on the foregoing, this BL is not applicable to the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 80-10, Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to Environment	This BL describes an event 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 stated that there are no permanent connections between the radioactive waste drain system (WRS) and nonradioactive piping. However, the design includes provisions 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 noncontaminated drainage system for disposal. WCAP-15800, Revision 3, states that this BL is not applicable to the AP1000 design certification and is the responsibility of the COL applicant. This is COL Action Item 20.7.1-1. The staff believes that such an event is not caused by poor system design but by 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. Based on the foregoing, this BL 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, Revision 3, Westinghouse stated that this BL 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. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 80-12, Decay Heat Removal Operability	This BL deals with reducing the likelihood of losing the decay heat removal capability in operating PWRs. In WCAP-15800, Revision 3, the applicant stated that DCD Section 7.4.1 addresses this BL. The AP1000 design relies on the passive RHR system for decay heat removal. For defense-in-depth considerations, the AP1000 design relies on the normal residual heat removal system (RNS) and associated procedures to reduce the shutdown mode risks. Sections 6.3 and 19.3 of this report discuss the staff evaluations of the PRHR capability and the shutdown risks involving RNS, respectively. Based on the staff's review, this BL is resolved for the AP1000 design.

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 does not fail 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 stated 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 MCR, the TSC, 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 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. The COL applicant will also address the emergency response facility communication system, including the crisis management radio system (DCD Tier 2, Section 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 certification. The reminder to the COL applicant to review this BL for recommendations related to loss of either onsite or offsite power, and a consequential loss of the ENS, is COL Action Item 20.7.4-1. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 80-18, Maintenance of Adequate Minimum Flow Through Centrifugal Charging Pumps Following Secondary-Side, High-Energy-Line Rupture	BL 80-18 recommends modification to equipment and/or procedures, if calculations determine the modification is necessary, to assure adequate minimum flow through the centrifugal charging pumps under all conditions. In WCAP-15800, Revision 3, Westinghouse stated that this BL is not applicable to the AP1000 design because the AP1000 design has no safety-related charging pumps as part of safety injection. The staff agrees. Based on the foregoing, this BL is not applicable to the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 80-20, Failure of the Applicant Type W-2 Spring Return to Neutral Control Switches	The applicant stated that this BL is not applicable to the AP1000 design certification because it involves a procurement issue. The staff agrees that the issues in this BL involve procurement and are the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 80-24, Prevention of Damage Due to Water Leakage Inside Containment	BL 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 CCW 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 is isolated by CIVs 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 ensures detection of an increasing sump water level. DCD Tier 2, Section 5.2.5, describes leakage detection, and DCD Tier 2, Section 3.4.1.2.2.1, addresses containment flooding events. Based on the staff's review, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 81-01, Revision 1, Surveillance of Mechanical Snubbers	The staff's 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 IST of pumps and valves and does not include any information on mechanical snubbers. The staff recognizes that BL 81-01 deals with examinations of snubbers installed in operating plants, and this aspect of the BL is not applicable to the AP1000 DC. However, the staff position is that the applicant should address the surveillance testing implications of this BL during the DC 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 of 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 ISI 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 Part 4 standard for IST of snubbers. DCD Tier 2, Section 3.9.8.3, states the requirement for the COL applicant to develop a program to verify the operability of snubbers utilized in the AP1000 design. This is COL Action Item 3.9.8-1. 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 IST plan for the AP1000 DC, and to Section XI of the ASME Code for performance of inservice inspection. WCAP-15800, Revision 3, provides 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's 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 BL by ensuring the establishment of programs for qualification testing of snubbers and inservice examination and functional testing of snubbers. Based on the staff's review, this BL 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, Revision 3, the applicant referenced 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 BL. The staff agrees with this basis. Because the subject of this BL led, in part, to the issuance of GL 89-10, the staff's position is that DCD Tier 2 should discuss the basis for disposition of BL 81-02. 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 MOVs provide an acceptable basis to resolve this issue. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL 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 BL is 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 BL is not applicable to the AP1000 design because the CCW and SWS do not cool safety-related components. In addition, DCD Tier 2, Section 9.2.1.2.2, addresses service water strainers and service water chemical injection. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 82-02, Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants	Westinghouse stated in WCAP-15800, Revision 3, that this BL is not applicable to the AP1000 design. Also, DCD Tier 2, Section 5.2.3.5, specifically prohibits the use of lubricants containing molybdenum disulfide in the AP1000 design. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 82-04, Deficiencies in Primary Containment Electrical Penetration Assemblies	This BL discusses the potential generic safety implications concerning electrical penetration assemblies supplied by the Bunker Remo Company. Westinghouse stated that this BL was not applicable to the AP1000 design certification because the issue involved procurement. The staff agrees that BL 82-04 will be resolved by the COL application. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 83-03, Check Valve Failures in Raw Water Cooling Systems of Diesel Generators	In WCAP-15800, Revision 3, Westinghouse stated that this issue is not applicable because the AP1000 DGs have no safety-related functions. The staff agrees with the applicant's assessment. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 84-03, Refueling Cavity Water Seal	This BL 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. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 85-02, Undervoltage Trip Attachments of the Applicant DB-50 Reactor Trip Breaker	Westinghouse stated in WCAP-15800, Revision 3, that it addresses this BL in DCD Tier 2, Section 7.1.2.1.2, and Chapter 16, TS, SR 3.3.1.5. This BL (1) assures proper reactor trip breaker (RTB) testing in plants that had not yet installed the automatic shunt trip modification and, (2) provides 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 BL. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 85-03, Motor-Operated Valve Common-Mode Failures during Plant Transients Due to Improper Switch Settings	In WCAP-15800, Revision 3, the applicant referenced DCD Tier 2, Section 3.9.6.2, as the basis for the resolution of this BL. The staff agrees with this basis. Because the subject of this BL led, in part, to the issuance of GL 89-10, the staff's position is that DCD Tier 2 should address the basis for the disposition of BL 85-03. 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 in service and qualification testing of MOV provides an acceptable basis to resolve this issue. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 86-01, Minimum Flow Logic Problems That Could Disable Residual Heat Removal (RHR) Pumps	This BL recommends that BWR licensees or applicants 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 miniflow lines for the pumps. In WCAP-15800, Revision 3, Westinghouse stated that this BL is not applicable to the AP1000 design because the AP1000 design has no valves in the miniflow lines for the normal RHR system, and the RHR pumps have no safety-related function. The staff agrees with the applicant's assessment. Based on the foregoing, this BL is not applicable for the AP1000 design.
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, Revision 3, that this BL is not applicable to the AP1000 design because the design does not have valves in miniflow lines. The staff has reviewed this issue and agrees with the applicant's assessment. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 87-01, Thinning of Pipe Walls in Nuclear Power Plants	In WCAP-15800, Revision 3, the applicant indicated that this BL is not applicable to the AP1000 design certification because it is a surveillance issue, as discussed in DCD Tier 2, Sections 5.4.3.4 and 10.3.6. This BL requests 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. It asks licensees 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 BL 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 that 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. This is COL Action Item 10.5-1. Based on the foregoing, the staff concludes that this BL 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 BL 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. The applicant stated that this issue is related to procurement and is not applicable to the AP1000 design certification in WCAP-15800, Revision 3. The NRC staff agrees with the applicant that BL 87-02 is not applicable to the AP1000 DC review since this is a procurement issue. The COL applicant is responsible for procurement issues. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 88-01, Defects of the Applicant Circuit Breakers	In WCAP-15800, Revision 3, the applicant stated that this issue is related to procurement and is not applicable to the AP1000 design certification. The NRC staff agrees with the applicant that BL 88-01 is not applicable to the AP1000 DC review since this is a procurement issue. The COL applicant is responsible for procurement issues. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes	Westinghouse stated that this BL was not applicable to the AP1000 design because there are stainless steel support plates in the AP1000 SGs. The staff agrees with this assessment. See also Section 5.4.2.1 of this report for additional information concerning AP1000 SGs. Based on the foregoing, this BL is not applicable to the AP1000 design.
BL 88-04, Potential Safety-Related Pump Loss	Westinghouse stated in WCAP-15800, Revision 3, that the BL is 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 concurs with this conclusion. Based on the foregoing, this BL 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, Revision 3, states that DCD Tier 2, Section 3.9.3.1.2 addresses this BL. 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 design. Section 3.12.5.9 of this report provides the staff's evaluation of this issue. Based on the staff's review, this BL is resolved for the AP1000 design.
BL 88-09, Thimble Tube Thinning in the Applicant Reactors	The staff reviewed the description of the design of the AP1000 thimble tubes given in DCD Tier 2, Section 3.9.7.2. The AP1000 thimble tubes have 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 nonisolable leak of reactor coolant. The staff's review concludes that the enhanced design of the AP1000 in-core instrumentation thimble tubes adequately addresses the BL 88-09 concerns for accelerated wear of in-core thimble tubes in the applicant's operating reactor designs. Based on the staff's review, this BL is resolved for the AP1000 design.
BL 88-11, Pressurizer Surge Line Thermal Stratification	WCAP-15800, Revision 3, states that DCD Tier 2, Section 3.9.3.1.2, addresses this BL. 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. Section 3.12.5.10 of this report provides the staff's evaluation of this issue. Based on the staff's review, this BL is resolved for the AP1000 design.
BL 89-01, Failure of the Applicant SG Tube Mechanical Plugs	In WCAP-15800, Revision 3, the applicant indicated that this BL is not applicable to the AP1000 design certification because the issue involves procurement. The staff agrees with this assessment since the COL applicant will purchase plugs installed into the SG following operation. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL 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, Revision 3, Westinghouse stated that this BL is not applicable to the AP1000 design certification because this is a procedural issue. The applicant also referred questions about this issue to DCD Tier 2, Sections 13.5.1, "Combined License Information Item," and 4.3.1.5, "Shutdown Margins." The staff agrees that this BL involves procedures; however, these procedures 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 maintain the required shutdown margin. DCD Tier 2, Section 9.1.6, also describes the responsibility of the COL applicant, which is designated as COL Action Item 20.7.4-2. Based on the foregoing, this BL is resolved for the AP1000.
BL 90-01, Loss of Fill-Oil in Transmitters Manufactured by Rosemount	The applicant stated in WCAP-15800, Revision 3, that this BL is not applicable to the AP1000 design certification because it involves a procurement issue. Supplement 1 to this BL states that transmitters manufactured after July 11, 1989, are not subject to the fill-oil leakage problems identified in the BL. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL 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. However, in WCAP-15800, Revision 3, Westinghouse also specified that this is a procurement issue, and, therefore, it is the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 93-02, Debris Plugging of Emergency Core Cooling Suction Strainers	This BL 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. However, in WCAP-15800, Revision 3, Westinghouse also specified that this is a procurement issue, and, therefore, it is the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL 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 BL 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 BL 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 CMTs 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 the following: The Combined License 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. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation consistent with COL Item 6.3.8.2. This is COL Action Item 6.2.1.8.1-1. Based on the foregoing, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 96-01, Control Rod Insertion Problems	This BL requires 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, Revision 3, Westinghouse stated that this BL is not applicable to the AP1000 design certification because it is a procedural issue. The IRI that led to the BLs 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 BL. Resolution of this issue is the responsibility of the COL applicant. This is COL Action Item 20.7.1-1. Based on the foregoing, this BL 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 BL reminds licensees of their responsibilities for ensuring safe performance of activities involving the movement of heavy loads. It also requests that licensees review their plans and capabilities for handling heavy loads and ensure that their load-handling operations are in accordance with existing regulatory guidelines and the licensing basis. The resolution of USI A-36 in DCD Tier 2, Section 1.9.4, addresses this issue. That section 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. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this BL 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 BL 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 BL 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 identified in this BL and considers it to be resolved for the AP1000 design. However, the staff is currently 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 the staff's evaluation of the design of the IRWST and containment recirculation screens in the context of Issue 191. This is COL Action Items 6.2.1.8.1-1 and 6.2.1.8.1-2. Based on the foregoing, this BL is resolved for the AP1000 design.
BL 2001-01, Circumferential Cracking of RPV Head Penetration Nozzles	In WCAP-15800, Revision 3, Westinghouse indicated that DCD Tier 2, Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," resolves this BL. This section of the DCD Tier 2 indicates that the use of Ni-Cr-Fe alloy in the AP1000 RCPB 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 find that this satisfactorily addresses all aspects of BL 2001-01. The COL applicant must perform additional inspections of the reactor vessel closure head and head penetrations. This is COL Action Item 4.5.1-1. As part of its review of DCD Tier 2, Section 4.5.1, "Control Rod Drive System Structural Materials," the staff considered the pertinent aspects of this BL as they apply to the design, fabrication, and inspection of control rod drive nozzle penetrations. Section 4.5.1 of this report contains the staff's evaluation of this information. Based on this evaluation, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	This BL sought information needed to determine the adequacy of PWR plants' Boric Acid Corrosion Control (BACC) Programs. It required all PWR addressees to submit to the NRC within 60 days the basis for concluding that their BACC Program for the RCPB provides reasonable assurance of compliance with the applicable regulatory requirements. The BL indicated that the staff would use the information submitted to determine the need for, and to guide the development of, additional regulatory actions to address degradation of the RPV head and/or other portions of the RCPB. Experience with currently operating PWRs continues to show cracking of Alloy 600 components. Recent experience indicates 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 the RCPB 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, the staff requested that Westinghouse provide information describing the extent to which the insulation of all Alloy 600/690 components and welds in the RCPB (not just upper reactor vessel head penetrations) will readily facilitate bare metal visual inspection during refueling outage conditions. This was Open Item 20.7-1 in the DSER. In a letter dated May 21, 2003, Westinghouse indicated that all the components with Alloy 690 type materials are accessible for inspection. For example, insulation can be removed for visual inspection, if necessary. Because these components are accessible for inspection through removal of insulation and because redesign of insulation to facilitate more rapid inspection, if necessary in the future, is not a major modification, the staff concludes that Westinghouse's response is acceptable, and on this basis, Open Item 20.7-1 is resolved. In addition, as noted in the BL, 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. Based on the foregoing, this BL is resolved for the AP1000 design.

Bulletin No. and Title	Staff Resolution
BL 2002-02, Reactor Pressure Vessel Head Penetration Nozzle Inspection Programs	This BL was issued after WCAP-15800, Revision 0. The staff's review of this application for BL 2002-01 relied on information contained in DCD Tier 2, Section 5.2.3. This section of DCD Tier 2 indicates that the use of Ni-Cr-Fe alloy in the AP1000 RCPB 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 find that this satisfactorily addresses all aspects of BL 2002-02. The COL holder must perform additional inspections of the reactor vessel closure head and head penetrations. This is COL Action Item 4.5.1-1. As part of its review of DCD Tier 2, Section 4.5.1, the staff considered the pertinent aspects of this bulletin as they apply to the design, fabrication, and inspection of control rod drive nozzle penetrations. Section 4.5.1 of this report contains the staff's evaluation of this information. Based on this evaluation, BL 2002-02 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

Generic Letter No. and Title	Staff Resolution
GL 80-01, Report on ECCS Cladding Models	GL 80-001 informs all licensees about an extension of 1 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 the AP1000 design. As indicated in WCAP-15800, Revision 3, WCAP-12945, “Westinghouse Code Qualification Document for Best Estimate Loss-of-Coolant Accident Analysis,” includes the NUREG-0630 cladding, swelling, and rupture models. DCD Tier 2, Chapter 15, addresses the LOCA analysis. Chapter 15 of this report contains the staff’s evaluation of the AP1000 LOCA analysis. Based on the foregoing, this issue is resolved for the AP1000 design.
GL 80-02, Quality Assurance Requirements Regarding Diesel Generator Fuel Oil	This GL concerns requirements on DG fuel oil in the QA program. In WCAP-15800, Revision 3, Westinghouse stated that this GL is not applicable to the AP1000 design, because the AP1000 does not have safety-related DGs, as discussed in DCD Tier 2, Section 8.3.1. The staff agrees. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 80-09, Low-Level Radioactive Waste Disposal	This GL concerns the requirements for solid waste shipments from a plant. To the extent that GL 80-09 applies to the design of the AP1000 design, DCD Tier 2, Section 11.4.2, addresses the requirements. In addition, to ensure that the COL applicant conforms to GL 80-09, 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. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 80-014, LWR Primary Coolant System Pressure Isolation Valves	In WCAP-15800, Revision 3, Westinghouse stated that DCD Tier 2, Section 1.9.4.2.2, Issue USI B-63, discusses this issue. The staff's 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. Section 3.9.3.1 of this report provides the staff's evaluation of this issue. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 80-019, Resolution of Enhanced Fission Gas Release Concern	In WCAP-15800, Revision 3, Westinghouse stated that this GL requires no action for the AP1000 design. However, 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," accounts for the fission gas release models for the AP1000 design. Based on the foregoing, this issue is resolved for the AP1000 design.
GL 80-026, Qualification of Reactor Operators	This GL set forth revised criteria for staff use in evaluating reactor operator training. Westinghouse stated that this GL is the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this issue is resolved for the AP1000 design.
GL 80-035, Effect of a dc Power Supply Failure on ECCS Performance	This GL addresses the concerns that the loss of a dc power supply could disable several ECCS components and thereby result in a limiting single-failure condition for some breaks. In WCAP-15800, Revision 3, Westinghouse stated that DCD Tier 2, Table 8.3.2-7, "Failure Modes and Effects Analysis," addresses this GL. The staff, in Section 8.3.2 of this report, has evaluated DCD Tier 2, Section 8.3.2 and concludes that DCD Tier 2, Table 8.3.2-7 adequately addresses the effect of a dc power supply on ECCS. Based on the staff's review, this GL 80-35 is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 80-077, Refueling Water Level	In WCAP-15800, Revision 3, Westinghouse stated that this GL is not applicable to the AP1000 design certification and was the responsibility of the COL applicant. DCD Tier 2, Sections 13.5.1 discuss this position. The staff agrees with the applicant that this issue is the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 80-088, Seismic Qualification of Auxiliary Feedwater Systems	WCAP-15800, Revision 3, states that this issue is not applicable to the nonseismic portion of the AP1000 startup feedwater system (SUFWS) (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 SUFWS is seismic Category I. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 80-109, Guidelines for SEP and Soil Structure Interaction Reviews	AP1000 is designed for hard rock sites only, as indicated in WCAP-15800, Revision 3. Section 3.7.2.4 of this report details the staff's review of the soil structure interaction (SSI) issue. The staff's review of this issue concludes that SSI is not applicable to the AP1000.
GL 81-014, Seismic Qualification of Auxiliary Feedwater Systems	WCAP-15800, Revision 3, states that this issue is not applicable to the nonseismic portion of the AP1000 SUFWS (inside the turbine building) and that the safety-related portion of this system in the containment and auxiliary building is seismically qualified (see DCD Tier 2, Section 10.4.9). The staff agrees with these safety classifications and concludes that the safety-related portion of the SUFWS is seismic Category I. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 81-021, Natural Circulation Cooldown	This GL addresses procedures and training to prevent, recognize, and react to reactor vessel voiding during natural circulation cooldown. In WCAP-15800, Revision 3, the applicant referred this GL to the ERGs. The staff reviewed the existing AP600 ERGs. In Chapter 18 of this report, the staff concluded that the AP600 ERGs are applicable to the AP1000 design, and that the operator has sufficient guidelines to cool down the plant using natural circulation means. Based on the staff's review, this GL is resolved for the AP1000 design.
GL 81-38, Storage of Low-Level Radioactive Wastes at Power Reactor Sites	This GL provides guidelines for the storage of low-level radioactive wastes at plant sites. The applicant stated in WCAP-15800, Revision 3, that this GL was not applicable to the AP1000 design certification because it is the responsibility of the COL applicant. This is a site-specific issue because it will depend on the available offsite storage space for low-level radioactive waste from the plant. The COL applicant will identify this issue 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 11.4.6, identifies GL 81-38 as a part of COL Action Item 11.4-1. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 81-39, NRC Volume Reduction Policy	This GL provides the Commission policy statement on reduction of low-level radioactive wastes at plant sites. DCD Tier 2, Section 11.4.2.1, addresses the application of GL 81-39 to the DC of the AP1000. 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. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 82-04, Use of INPO (Institute of Nuclear Power Operations) SEE-IN Program	This GL recommends 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 discussed the review of operating experience in the resolution of TMI Action Plan Item I.C.5 in DCD Tier 2, Chapters 1 and 18. The staff found this discussion acceptable. Further, in WCAP-15800, Revision 3, Westinghouse specified that this issue is the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this 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 GL is not applicable to the AP1000 design certification because it is an administrative communication to the licensees. This GL is not a design issue because site security is the responsibility of the COL applicant. This includes the reporting of safeguards information for licensing reviews. Based on the foregoing, the GL is not applicable to the AP1000 design.
GL 83-11, Licensee Qualifications for Performing Safety Analyses in Support of Licensing Actions	This GL provides 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, Revision 3, the applicant stated 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's review, this GL is resolved for the AP1000 design. However, the staff identifies in COL Action Item 20.7.4-3 that if a COL applicant chooses to perform its own safety analysis in the future, it will follow the guidelines specified in GL 83-11, Supplement 1.

Generic Letter No. and Title	Staff Resolution
GL 83-14, Definition of Key Maintenance Personnel	In WCAP-15800, Revision 3, the applicant stated that this GL was not applicable to the AP1000 design because it is an administrative communication. The staff agrees. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 83-15, Implementation of RG 1.150, “Ultrasonic Testing of Reactor Vessel Welds During Preservice and In service Testing”	GL 83-15 was issued to inform applicants and licensees to use the ultrasonic testing methods in RG 1.150, Revision 1, for pre-service and inservice inspections, except in those cases in which an alternative method is proposed for complying with the Commission’s Regulations. The applicant stated in DCD Tier 2, Appendix 1A, that the AP1000 design conforms to the recommendations of RG 1.150, therefore, this GL is resolved for the AP1000 design.
GL 83-21, Clarification of Access Control for Law Enforcement Visits	The completion of the security review was Open Item 13.6-1 in the DSER. The resolution of Open Item 13.6-1 is contained in Chapter 13.6 of this report. The staff reviewed WCAP-15800, Revision 3, and concluded that GL 83-21 was adequately addressed by the applicant. The issues associated with this GL will be further addressed by the COL applicant. This is COL Action Item 13.6-1. Based on the staff’s review, this GL is resolved for the AP1000 design.
GL 83-22, Safety Evaluation of “Emergency Response Guidelines”	This GL states that the applicant’s ERG program was acceptable and provided improved guidance for development of plant EOPs. In WCAP-15800, Revision 3, Westinghouse stated that this GL is not applicable to the AP1000 design certification because it is the responsibility of the COL applicant. The staff reviewed the applicant’s ERG program and documented its evaluation in Section 18.9.3 of this report. The staff also identified COL Action Item 18.9.3-1 for the COL applicant to develop plant-specific EOPs using the ERGs. Based on the staff’s review, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 83-32, NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS	The applicant stated that DCD Tier 2, Section 18.8.2, addresses this GL. The staff has completed its review of DCD Tier 2, Section 18.8.2, and finds the applicant's position acceptable, as discussed in Section 18.8 of this report. Based on the staff's 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	DCD Tier 2, Section 9.5.1, addresses this GL. In reviewing the AP1000 design, the staff considered the positions of the GL, and based on the evaluation in Section 9.5.1 of this report, this GL is resolved for the AP1000 design.
GL 83-41, Fast Cold Start of Diesel Generator	The applicant stated in DCD Tier 2, Section 8.3, that DGs for the AP1000 design are not safety-related. Therefore, this GL is not applicable to the AP1000 design. The staff agrees. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 84-04, Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops	WCAP-15800, Revision 3, references DCD Tier 2, Section 1.9.4.2.2, USI A-2, for the response to this issue. The staff's 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 also referred to as LBB 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's review of the applicant's LBB criteria appears in Section 3.6.3 of this report. Based on the staff's review, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 84-09, Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii)	This issue was a DSER open item because DCD Tier 2 did not comply with the current regulations for the control of combustible gas in containment during accidents. This was Open Item 6.2.5-1 in the DSER. Subsequent to the publication of the DSER, the NRC revised its regulations regarding the control of combustible gas in containment. The revised regulations were published on September 16, 2003, and became effective on October 16, 2003. The NRC has extensively revised 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” made associated changes to Subsections 50.34 and 52.47, and added a new section, Subsection 50.46a, “Acceptance Criteria for Reactor Coolant System Venting Systems.” The revisions apply to current power reactor licensees, and also consolidate combustible gas control regulations for future power reactor applicants and licensees. The revised rules eliminate the requirements for hydrogen recombiners and hydrogen purge systems, and relax the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance. Because the revised rules have generically eliminated the hydrogen recombiner requirements, this generic letter is does not apply to the AP1000 design and Open Item 6.2.5-1 is closed. Therefore, based on the foregoing, this GL is resolved for the AP1000 design.
GL 84-12, Compliance with 10 CFR Part 61 and Implementation of Radiological Effluent TS, Attendant Process Control Program	This GL addresses a concern about compliance with 10 CFR Part 61 and implementation of the radiological effluent TS, Attendant Process Control Program. GL 84-12 has been superseded by GL 89-01, which has been incorporated into TS 5.5.2, “Radioactive Effluent Control Program,” in a manner consistent with the guidance provided in NUREG-1431. In addition, DCD Tier 2, Section 11.4.6, refers to 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 Sections 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 foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 84-15, Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	This GL addresses the reliability of the DG, which has been identified as being one of the main factors affecting the risk from SBO. Thus, attaining and maintaining high reliability of DGs were necessary to the resolution of USI A-44. As stated by the applicant in DCD Tier 2, Section 1.9.1, and Appendix 1A, concerning the inapplicability of RG 1.108, this GL is not applicable because the DGs in the AP1000 design are not safety-related and are not required for accident mitigation. The staff agrees. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 84-21, Long-Term, Low-Power Operation in PWRs	This GL is 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, Revision 3, Westinghouse stated that this GL is not applicable to the AP1000 design certification because it is 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. Chapter 15 of this report discusses the safety evaluation of this issue. Based on the staff's review, this GL is resolved for the AP1000 design.
GL 85-05, Inadvertent Boron Dilution Events	GL 85-05 informs each PWR licensee of the staff position resulting from the evaluation of Issue 22, and urges each licensee to ensure that its plants have adequate protection against boron dilution events. In WCAP-15800, Revision 3, the applicant referred this issue to DCD Tier 2, Section 15.4.6. The staff evaluated and discussed this issue in Chapter 15 of this report. To mitigate the consequence of this event, operator actions must isolate the potential unborated water from the demineralized water transfer and storage system, or CVS. The staff states in COL Action Item 20.7.4-4 that the COL applicants should develop plant-specific EOPs that address the boron dilution events. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 85-06, Quality Assurance Guidance for ATWS Equipment That is Not Safety Related	10 CFR 50.62(c)(1) states, in part, that each PWR must have equipment from sensor output to final actuation device, that is diverse from the RTS, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS event. GL 85-06 provides the explicit QA guidance required by 10 CFR 50.62, “Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants,” for the non-safety related SSCs required to mitigate an ATWS event per 10 CFR 50.62(c)(1). During the evaluation of the applicant’s resolution of this generic issue, the NRC staff reviewed 10 CFR 50.62; the QA guidance contained in GL 85-06; AP1000 DCD Sections 15.8, “Anticipated Transients Without Scram,” 17.3, “Quality Assurance,” Table 17-1, “Quality Assurance Program Requirements for Systems, Structures and Components Important to Investment Protection;” and WCAP-15985, Revision 1, “AP1000 Implementation of the Regulatory Treatment of Non-Safety Systems Process,” dated April 2003. In AP1000 DCD Section 15.8, the applicant stated that the AP1000 DAS provides the ATWS mitigation systems actuation circuitry protection features mandated for Westinghouse plants by 10 CFR 50.62. For ATWS mitigation, the DAS trips the turbine and actuates PRHR to provide decay heat removal for the AP1000. In DCD Section 7.7.2.11, “Diverse Actuation System,” 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, “Diverse Actuation System,” of this report. The applicant addressed QA requirements for the SSCs providing ATWS mitigation under the RTNSS process described in SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Designs.” In WCAP-15985, the applicant provided the proposed resolution for the AP1000 RTNSS policy issue. Section 4.1 of WCAP-15985 states that the DAS functions of reactor trip, turbine trip, and PRHR actuation functions, and the associated non-Class 1E dc and UPS system power supplies, are needed to meet the requirements of 10 CFR 50.62. Section 10.3.1 of WCAP-15985 states that the QA guidance provided in GL 85-06 is

Generic Letter No. and Title	Staff Resolution
GL 85-06, Continued	applicable to the DAS. However, the staff identified that Section 10.3.3 of WCAP-15985 did not include a similar requirement to apply the GL 85-06 quality control guidance to the non-Class 1E and UPS power systems that support the DAS ATWS functions. The staff concluded that the non-Class 1E and UPS power systems that support the DAS ATWS functions are required to mitigate an ATWS event in accordance with 10 CFR 50.62 and should be subject to the GL 85-06 QA guidance. The staff also noted that DCD Tier 2, Section 17.3 states that quality requirements for systems, structures, and components included in the RTNSS are identified in DCD Tier 2, Table 17-1. Furthermore, DCD Tier 2, Section 17.3 references DCD Tier 2, Section 16.3, “Investment Protection,” for systems that should be considered for designation of systems and components included in the RTNSS. The staff noted that the DAS system and the non-Class 1E dc and UPS systems that provide power to DAS are included within the scope of DCD Tier 2, Section 16.3. Additionally, the DAS manual controls are specified in the AP1000 TS 3.3.5 (DCD Chapter 16). Therefore, the staff concluded that the DAS and the associated non-Class 1E dc and UPS support systems are subject to the DCD Tier 2, Table 17-1 QA controls. The staff reviewed DCD Tier 2, Table 17-1 QA controls and determined that the QA controls specified in DCD Tier 2, Table 17-1 adequately addressed quality assurance guidance for non-safety related ATWS equipment in GL 85-06. In comparing the information contained in the AP1000 DCD and WCAP-15985, the staff noted that the QA controls specified for the ATWS mitigation equipment required by 10 CFR 50.62 appeared to be inconsistent. Specifically, the WCAP-15985 does not specify quality assurance requirements for the non-Class 1E and UPS systems, while the DCD indicates that these systems are subject to quality assurance guidance equivalent to GL 85-06. Therefore, the staff determined that the applicant should clearly state the QA 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 was identified as DSER Open Item 20.7-2. In August 2003, WCAP-15985, Revision 2, Westinghouse states in Section 10.3-1, “Instrumentation Systems,” that the QA guidance

Generic Letter No. and Title	Staff Resolution
GL 85-06, Continued	provided in GL 85-06 is applicable to DAS because of the ATWS mitigation functions (DCD Tier 2, Section 7.7.1.11). Westinghouse also states in DCD Tier 2, Section 10.3.3, “Electrical Systems,” that the QA guidance provided in GL 85-06 is applicable to the non-Class IE, dc and UPS systems because of the ATWS mitigation functions (DCD Tier 2, Section 8.2.3.1.2). The NRC staff found that this adequately addresses QA requirements for DAS, the non-Class 1E, dc and UPS systems for the purposes of satisfying the requirements of GL 85-06; therefore, Open Item 20.7-2 is resolved. Based on the staff’s review, this GL is resolved for the AP1000 design.
GL 85-13, Transmittal of NUREG-1154 Regarding the Davis-Besse Loss of Main and Auxiliary Feedwater Event	NUREG-1154 addresses the loss of all feedwater event on June 9, 1985 at the Davis-Besse plant. The causes of this event were (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 failure of the licensee’s evaluation of operating experience related to equipment to always find the root causes of problems and correct them, 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 an inadequate system maintenance program caused the Davis-Besse event. The AP1000 does not have an AFWS. The SUFWS, 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 PXS, including the PRHR HXs, is the safety-related means of providing emergency cooling for the RCS. The applicant addressed the Davis-Besse event in its plant operating and maintenance procedures for the main and startup feedwater systems. DCD Tier 2, Section 13.5.1, identifies the COL applicant as having the responsibility for the preparation of plant operating procedures. The staff agrees with the applicant’s assessment. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 85-16, High Boron Concentrations	This GL encourages licensees to reevaluate the need for high boron concentration (about 20,000 ppm boron) in the boron injection tank. In WCAP-15800, Revision 3, 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 has only a CMT with a maximum boron concentration of 3300 ppm boron. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 86-04, Policy Statement on Engineering Expertise on Shift	The applicant has satisfactorily addressed this GL in DCD Tier 2, Section 18.6, "Staffing," and has identified it as a COL responsibility. This is COL Action Item 18.6.3-1. Based on the foregoing, 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 transmits incident investigation report NUREG-1190 and requests licensees to review the report for applicability to their facility. In WCAP-15800, Revision 3, the applicant stated that this GL is resolved in Issue A-1 in DCD Tier 2, Section 1.9.4.2.2. The staff agrees. Section 20.2 of this report discusses the staff's evaluation of Issue A-1. Based on the staff's review, this GL is resolved for the AP1000 design.
GL 86-10, Implementation of Fire Protection Requirements	This GL provides guidance on meeting 10 CFR Part 50, Appendix R, which superseded 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's review, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 86-16, The Applicant ECCS Evaluation Models	This GL concerns the need for additions and corrections to the applicant's ECCS evaluation models that contain the WREFLOOD and BART computer codes. In WCAP-15800, Revision 3, the applicant referred 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 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 AP1000 design does not include the two computer codes referred to in GL 86-16. 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 foregoing, 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 3, the applicant stated that it has addressed this GL in DCD Tier 2, Chapter 16 TS, LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Integrity." Section 3.9.6.2 of this report discusses the staff evaluation of this issue. The staff concludes that DCD Tier 2, Table 3.9-18, contains an acceptable list of PIVs, and LCO 3.4.15 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 this GL is resolved for the AP1000 design.
GL 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements	This GL issues Revision 2 to BTP Mechanical Engineering Branch (MEB) 3-1 of SRP 3.6.2 to eliminate the guidelines for postulating arbitrary intermediate pipe ruptures. The applicant stated that it has addressed this GL in DCD Tier 2, Section 3.6.2. DCD Tier 2, Section 3.6.2, provides information relative to postulating pipe ruptures that is consistent with BTP MEB 3-1, Revision 2. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
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 requests licensees to provide information regarding midloop operation, and GL 88-17 provides guidance to licensees. In WCAP-15800, Revision 3, Westinghouse stated that it has addressed this GL in DCD Tier 2, Section 1.9.5.1, SECY-90-016 Issues. In DCD Tier 2, Section 1.9.5.1.4, the applicant discussed the AP1000 design's compliance with the issue of midloop operation. Section 19.3 of this report discusses the staff's resolution of this issue. Based on the staff's 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, "Severe Accidents," of this report on severe accidents and PRA for the design. Based on the staff's review, this GL is resolved for the AP1000 design.
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 3, the applicant indicated that this GL is not applicable to the AP1000 design certification and 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. As stated in DCD Tier 2, Section 5.2.6.2, the AP1000 COL applicant will develop a Boric Acid Corrosion Program to provide reasonable assurance of compliance with the applicable regulatory requirements. This is part of COL Action Item 20.7.4-5. Based on the foregoing, GL 88-05 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 it has addressed this GL in Section 1.9.4.2.2 and Appendix 1A as it involves Issue A-47 and RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. Section 20.2 of this report discusses Issue A-47, which deals with safety implications of control systems. Discussion of Issues A-11 and 15, which involve reactor vessel materials and radiation, appear in Sections 20.2 and 20.3, respectively, of this report. Based on the staff's review, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment	Westinghouse stated that it has addressed this GL in DCD Tier 2, Section 9.3.1. Section 20.3 of this report provides staff resolution of Issue 43. Based on the staff's 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. These problems include onsite distribution system voltages lower than required for proper operation of safety equipment, DG loading exceeding design, inadequate DG response to actual loading, overloading Class 1E buses, inadequate breaker coordination, and inadequate fault current interruption capability. For GL 88-15, Westinghouse referred to DCD Tier 2, Section 8.3.1.1.2.1, in WCAP-15800, Revision 3. The investment protection short-term availability controls described in DCD Tier 2, Section 16.3, and the D-RAP described in DCD Tier 2, Section 17.4, include the standby DGs. The inspection, test, analyses, and acceptance criteria (ITAAC) cover the breaker coordination and fault current interruption capability. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 88-17, Loss of Decay Heat Removal	This GL concerns loss of decay heat removal during nonpower operation. In WCAP-15800, Revision 3, Westinghouse stated that it has discussed this GL in DCD Tier 2, Section 1.9.5.1. DCD Tier 2, Section 1.9.5.1.4, discusses this GL and GL 87-12 in midloop operation for SECY-90-016 issues. (The SRM to SECY-90-016 provides four additional recommendations for decay heat removal during midloop operation.) Section 19.3 of this report discusses the staff's resolution of this issue. Based on the staff's review, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 88-19, Use of Deadly Force by Licensee Guards to Prevent Theft of Special Nuclear Material	The completion of the security review was Open Item 13.6-1 in the DSER. The resolution of Open Item 13.6-1 is contained in Section 13.6 of this report. The issues associated with this GL will be further addressed by the COL applicant. This is COL Action Item 13.6-1. Based on the foregoing, this GL is resolved for the AP1000 design.
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's review, this GL is resolved for the AP1000 design.
GL 89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products	The purpose of GL 89-02 is to share with all licensees some of the elements of programs that appear to be effective in detecting counterfeit or fraudulently marketed products and in assuring the quality of vendor products. In WCAP-15800, Revision 3, the applicant stated that this issue is related to procurement and is not applicable to the AP1000 design certification. The NRC staff agrees with the applicant that GL 89-02 is not applicable to the AP1000 DC review since this is a procurement issue, which is the responsibility of the COL applicant. This is COL Action Item 20.7.1-1. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 89-04, Guidance on Developing Acceptable Inservice Testing Programs	In WCAP-15800, Revision 3, Westinghouse stated that DCD Tier 2, Section 3.9.6.2 addresses this GL. The staff based its evaluation and acceptance of the AP1000 IST program on the information in DCD Tier 2, Section 3.9.6. Section 3.9.6 of this report includes the details of the staff's review of the AP1000 IST program. Based on the staff's review, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 89-07, Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	The completion of the security review was Open Item 13.6-1 in the DSER. The resolution of Open Item 13.6-1 is contained in Chapter 13.6 of this report. The issues associated with this GL will be further addressed by the COL applicant. This is COL Action Item 13.6-1. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 89-08, Erosion/Corrosion Induced Pipe Wall Thinning	This GL requests information on the long-term erosion/corrosion monitoring program to ensure the maintenance of the structural integrity of all high-energy carbon steel systems. The applicant stated that this GL is a surveillance issue, which it discusses 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 prepare an erosion/corrosion surveillance program using industry guidelines. This is COL Action Item 10.5-1. The staff agrees with this assessment. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance GL 89-10 S1, Results of the Public Workshops GL 89-10 S5, Inaccuracy of Motor-Operated Valve Diagnostic Equipment GL 89-10 S6, Information on Scheduling and Grouping and Staff Responses to Additional Public Questions GL 89-10 S7, Consideration of Valve Mispositioning in PWRs	WCAP-15800, Revision 3, 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's 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 MOVs, provides an acceptable basis for resolution of GL 89-10 for the AP1000 design. This is part of COL Action Item 20.7.1-1. Based on the staff's review, this GL is resolved for the AP1000 design.
GL 89-13, Service Water System Problems Affecting Safety-Related Systems	This GL requests information about the compliance of SWSs with certain GDC and quality assurance requirements, such as test control. The applicant stated in WCAP-15800, Revision 3, that it does not use the SWS for safety-related cooling in the AP1000. Therefore, this GL is not applicable to the AP1000. The staff agrees with the applicant's assessment. Based on the foregoing, this GL is not applicable to the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 89-15, Emergency Response Data System	DCD Tier 2, Section 13.3, “Emergency Planning”, states that emergency planning, including communication interfaces among the MCR, the TSC, and the emergency planning centers, is the responsibility of the COL applicant. The COL applicant referencing the AP1000 certified design will address emergency planning, including post-72-hour actions and 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. The COL applicant will also address the emergency response facility communication system, including the crisis management radio system. Therefore, the staff concludes that this GL is not applicable to the AP1000 design certification, and that it is the responsibility of the COL applicant. The staff notes, however, that 10 CFR Part 50, Appendix E, Section VI.2(a)(i), provides the selected emergency response data system (ERDS) plant parameters for PWRs. Because of the unique design of the AP1000, the plant parameters required for the ERDS will be similar, but not all inclusive. The COL applicant must review this GL and 10 CFR Part 50, Appendix E, to ensure that the necessary AP1000 plant parameters are available to the ERDS. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 91-05, License Commercial Grade Procurement and Dedication Programs	The purpose of GL 91-05 is to allow licensees sufficient time to fully understand and implement guidance developed by industry to improve procurement and commercial grade dedication programs. In WCAP-15800, Revision 3, the applicant stated that this issue is related to procurement and is not applicable to the AP1000 design certification. The NRC staff agrees with the applicant that GL 91-05 is not applicable to the AP1000 DC review because this is a procurement issue and the COL applicant is responsible for procurement issues. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 91-07, GSI-23, “RCP Seal Failure” and Its Possible Effect on SBO	GL 91-07 informs licensees of the possible effect of GSI 23 on their responses to the SBO rule. In WCAP-15800, Revision 3, Westinghouse stated that this GL is not applicable to the AP1000 design because the design of the AP1000 reactor coolant pumps obviates the need for reactor coolant pump seals, as discussed in DCD Tier 2, Sections 5.1.3.3 and 1.9.4.2.3. The staff agrees with the applicant’s assessment. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 91-14, Emergency Telecommunications	This GL alerts reactor power plant licensees to the NRC’s effort to upgrade its emergency telecommunications system. The NRC identified seven essential communications functions and requested licensees to modify 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, states that emergency planning, including communication interfaces among the MCR, the TSC, and the emergency planning centers, is the responsibility of the COL applicant. The COL applicant referencing the AP1000 certified design will address emergency planning, including post-72-hour actions and 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. The COL applicant will also address the emergency response facility communication system, including the crisis management radio system. The staff considers this issue to be outside the scope of the AP1000 DC and, therefore, the COL applicant will address it. DCD Tier 2, Section 13.3, covers this issue and DCD Tier 2, Section 13.3.2, designates it as COL Information Item 13.3-1. The COL applicant will review the guidance in the GL associated with emergency telecommunications. This is part of COL Action Item 20.7.4-6. Based on the foregoing, GL 91-14 is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 91-15, Operating Experience Feedback Report, Solenoid-Operated Valve Problems at U.S. Reactors	This GL informs licensees of a case study on solenoid-operated valves by AEOD but does not request any specific action. The applicant stated that this GL is not applicable to the AP1000 design certification because it involves 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. This is a part of COL Action Item 20.4-2. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 91-16, Licensed Operators' and Other Nuclear Facility Personnel's Fitness for Duty	The completion of the security review was Open Item 13.6-1 in the DSER. The resolution of Open Item 13.6-1 is contained in Chapter 13.6 of this report. The issues associated with this GL will be further addressed by the COL applicant. This is COL Action Item 13.6-1. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 92-01, Revision 1, Reactor Vessel Structural Integrity, and GL 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity	GL 92-01, Revision 1, Supplement 1, requests all licensees to review their RPV structural integrity assessments to identify, collect, and report any new data pertinent to the analysis of the structural integrity of their RPVs and to assess the impact of that data on their RPV integrity analyses. Similarly, the staff is requesting that applicants using the AP1000 design, as part of their actual plant submittal, provide the amount of copper, nickel, and phosphorus contents, the initial RT_{NDT} value, the projected fluence at the end of the license period for the limiting material, and the method of calculating the fluence. This information will confirm that the proposed P/T limits are in accordance with 10 CFR Part 50, Appendix G, and that the beltline materials conform to the PTS criteria of 10 CFR 50.61. By letter dated November 7, 2003, Westinghouse indicated that the COL applicant will address verification of plant-specific beltline material properties consistent with DCD Tier 2, Section 5.3.3.1 concerning P/T limit curves; DCD Tier 2, Table 5.3-1 concerning the maximum limits for elements of the reactor vessel; and DCD Tier 2, Table 5.3-3 concerning end-of-life RT_{NDT} and upper shelf energy projections. The verification will include a PTS evaluation and an upper shelf energy evaluation based on as-procured vessel material data and the projected neutron fluences for the plant design objective of 60-years. The COL applicant will submit these evaluation reports for NRC staff review. The PTS evaluation will include the copper, nickel, and phosphorus contents, the initial RT_{NDT} value, the projected fluence at the end of the license period for the limiting material, and the method of calculating the fluence. The upper shelf energy evaluation will include the initial upper shelf energy and the projection of upper shelf energy to the end of life based on copper content and the projected fluence. Subsequent to the November 7, 2003, letter, Westinghouse added this COL commitment to DCD Tier 2, Section 5.3.6.4. This commitment is responsive to the information the staff requested under GL 92-01, Revision 1, Supplement 1. This is COL Action Item 20.7.4-7. Based on the foregoing, GL 92-01, Revision 1, Supplement 1, is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 93-01, Emergency Response Data System Test Program	DCD Tier 2, Section 13.3, states that emergency planning, including communication interfaces among the MCR, the TSC, and the emergency planning centers, is the responsibility of the COL applicant. The COL applicant referencing the AP1000 certified design will address emergency planning, including post-72-hour actions and 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. The COL applicant will also address the emergency response facility communication system, including the crisis management radio system. This is COL Action Item 20.7.4-8. The staff agrees that this issue is the responsibility of the COL applicant. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 93-04, Rod Control System Failure and Withdrawal of Rod Cluster Assemblies	This GL addresses the single-failure vulnerability within the applicant's 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, Revision 3, Westinghouse referred to DCD Tier 2, Section 3.9.4, "Control Rod Drive System," to address this issue. WCAP-13864, Revision 1-A, "Rod Control System Evaluation Program," provided Westinghouse 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, Revision 1, and requires preoperational and startup testing as specified in DCD Tier 2, Sections 14.2.9.1.8 and 14.2.10.1.11. The COL applicant will perform additional testing during the operational phase of the plant. This is COL Action Item 20.7.4-9. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Issues

Generic Letter No. and Title	Staff Resolution
GL 94-01, Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators	The applicant stated in DCD Tier 2, Section 1.9.4, Item B-56, that this GL is not applicable to the AP1000 design. The onsite ac electrical power sources, the DGs, are non-safety-related in the AP1000 design, and 10 CFR 50.36 does not require them to be in the TS. Thus, this GL does not apply to the AP1000 TS. Based on the foregoing, this GL is not applicable to 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 through Issue A-3. The staff's review of Issue A-3 is documented in Section 20.2 of this report, and Issue A-3 is resolved for the AP1000 design. Based on the staff's review, this GL is resolved for the AP1000 design.
GL 95-07, Pressure Locking and Thermal Binding of Safety-Related, Power-Operated Gate Valves	This GL requests all holders of OLs or CPs 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 to take necessary corrective actions to ensure operability of the applicable valves. For the AP1000 DC, 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 3, references DCD Tier 2, Sections 5.4.8.1.2 and 5.4.8.2, for resolution of this GL. The staff's 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. WCAP-15800, Revision 3, specifies that this is also a procurement issue, and, therefore, is the responsibility of the COL applicant. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 96-01, Testing of Safety-Related Logic Circuits	This GL addresses 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 needed. The applicant addressed testing of safety-related logic circuits in DCD Tier 2, Section 7.1.2 and 13.5. However, comparing electrical schematic drawings and logic diagrams against plant surveillance test procedures to ensure that the surveillance procedures fulfill the TS requirements is the responsibility of the COL applicant. This is COL Action Item 20.7.4-10. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 96-02, Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat	The completion of the security review was Open Item 13.6-1 in the DSER. The resolution of Open Item 13.6-1 is contained in Chapter 13.6 of this report. The issues associated with this GL will be further addressed by the COL applicant. This is COL Action Item 13.6-1. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 96-04, Boraflex Degradation in Spent Fuel Pool Storage Racks	This GL asks licensees who use Boraflex as a neutron absorber in their 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, Revision 3, the applicant stated that this is a procurement issue and is not within its scope of the AP1000 design certification. The staff agrees. 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-percent subcriticality margin as described in DCD Tier 2, Section 9.1.6. This is COL Action Item 20.7.4-11. Based on the foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 96-05, Periodic Verification of Design Basis Capability of Safety-Related Motor Operated Valves	WCAP-15800, Revision 3, 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 IST program consistent with the recommendations in GL 89-10 and its supplements, as well as in 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 for IST in conformance with DCD Tier 2, Section 3.9.6 and Table 3.9-16, and in situ testing to confirm the capacity of the valve to operate under design conditions. This is part of COL Action Item 3.9.6.4-1. The staff's evaluation of this issue is contained in Section 3.9.6 of this report. Based on its review of this information, the staff concludes that GL 96-05 is resolved for the AP1000 design.
GL 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	This GL addresses concerns associated with water hammer, two-phase flow, and thermally induced overpressurization. The GL requests that licensees evaluate systems that are 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 requirements in 10 CFR Part 50, Appendix B. If corrective actions are required, the GL reminds licensees of their responsibility to ensure that systems remain operable and can continue to perform their safety functions while the corrective actions are being implemented. In WCAP-15800, Revision 3, 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 subsystems are not safety-related. Therefore, GL 96-06 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 foregoing, this GL is resolved for the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 97-01, Degradation of CRDM Nozzle and Other Vessel Closure Head Penetrations	BLs 2001-01 and 2002-02 have superseded GL 97-01. See the resolution to BLs 2001-01 and 2002-02 in Table 20.7-1 of this report and DCD Tier 2, Section 5.2.3.1.
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 ECCS pumps. Because the AP1000 does not use ECCS pumps, this GL is not applicable to the AP1000 design. Based on the foregoing, this GL is not applicable to the AP1000 design.
GL 97-05, Steam Generator Tube Inspection Techniques	WCAP-15800, Revision 3, states that this GL is not applicable to the AP1000 design certification. The staff agrees because this GL relates to SG tube inspections conducted in accordance with approved inspection procedures, and, as such, it is outside the scope of DC. This is part of COL Action Item 20.7.1-1. Based on the foregoing, this GL is not applicable to the AP1000 design.

Generic Letter No. and Title	Staff Resolution
GL 97-06, Degradation of Steam Generator Internals	This GL requests, in part, that licensees discuss any programs in place to detect degradation of SG internals and include a description of the plans, scope, frequency, methods, and equipment used. In WCAP-15800, Revision 3, the applicant indicated that this GL is not applicable to the AP1000 design certification because it is a procedural issue and the tube supports are fabricated from stainless steel. The staff agrees that this is a procedural issue 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 SG internals. As a result, the staff concludes that the COL applicant will need to develop a program for periodic monitoring for potential degradation of SG internals. Subsequent to the issuance of the DSER, Westinghouse added a commitment to DCD Tier 2, Section 5.4.1.5, indicating that the COL applicant will develop a program for periodic monitoring for potential degradation of SG internals. The staff finds this commitment acceptable as it addresses the staff's concern regarding a monitoring program for potential degradation of SG internals. This is COL Action Item 20.7.4-12. Based on the foregoing, this GL is resolved for the AP1000 design.
GL 98-02, Loss of Reactor Coolant Inventory and Associated Potential Loss of Emergency Mitigation Functions While in Shutdown Condition	GL 98-02 requests PWR licensees to assess the susceptibility of their RHR and ECCS to common-cause failure as a result of RCS draindown in a shutdown condition. In WCAP-15800, Revision 3, Westinghouse stated that this issue is not applicable to the AP1000 because the 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 rely 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 foregoing, this GL is not applicable the AP1000 design.

Generic Letter 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 addresses 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 the following: The Combined License 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. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation consistent with COL Item 6.3.8.2. This is COL Action Item 6.2.1.8.1-1. In DCD Tier 2, Section 6.1.3.2, the applicant addressed programmatic controls to ensure the proper procurement, application, and monitoring of safety-related coatings for the AP1000. Therefore, the staff concludes that the applicant has resolved two of the three main issues raised in this GL. With respect to the third issue of non-safety-related or unqualified coatings, GL 98-04 offers the option of demonstrating 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 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal	DCD Tier 2, Sections 6.4, 9.4.1, and 9.4.7, states that the AP1000 design has no safety-related filtration systems. The NRC staff agrees with the applicant that RG 1.52, and GL 99-02 are not applicable to the AP1000 design. Based on the foregoing, this GL is not applicable to the AP1000 design.