CHAPTER 6

ENGINEERED SAFETY FEATURES

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CHAPTER 6

ENGINEERED SAFETY FEATURES

6.0 ENGINEERED SAFETY FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.1.1.2 Fabrication Requirements

Add the following information to the end of DCD Subsection 6.1.1.2:

STD COL 6.1-1

In accordance with Appendix B to 10 CFR Part 50, the quality assurance program establishes measures to provide control of special processes. One element of control is the review and acceptance of vendor procedures that pertain to the fabrication, welding, and other quality assurance methods for safety related component to determine both code and regulatory conformance. Included in this review and acceptance process are those vendor procedures necessary to provide conformance with the requirements of Regulatory Guides 1.31 and 1.44 for engineered safety features components as discussed in DCD Section 6.1 and reactor coolant system components as discussed in DCD Subsection 5.2.3.

6.1.2.1.6 Quality Assurance Features

Replace the third paragraph under the subsection titled “Service Level I and Service Level III Coatings” within DCD Subsection 6.1.2.1.6 with the following information.

STD COL 6.1-2

During the design and construction phase, the coatings program associated with selection, procurement and application of safety related coatings is performed to applicable quality standards. The requirements for the coatings program are contained in certified drawings and/or standards and specifications controlling the coating processes of the designer (Westinghouse) (these design documents will be available prior to the procurement and application of the coating material by the constructor of the plant). Regulatory Guide 1.54 and ASTM D5144 (Reference 201) form the basis for the coatings program.

During the operations phase, the coatings program is administratively controlled in accordance with the quality assurance program implemented to satisfy 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements. The coatings program provides direction for the procurement, application, inspection, and monitoring of safety related coating systems. Prior to initial fuel loading, a consolidated plant coatings program will be in place to address procurement, application, and monitoring (maintenance) of those coating system(s) for the life of the plant.

Coating system monitoring requirements for the containment coating systems are based on ASTM D5163 (Reference 202), "Standard Guide for Establishing

Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant," and ASTM D7167 (Reference 203), "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality assurance requirements.

Include a new second paragraph under the subsection titled "Service Level II Coatings" within DCD Subsection 6.1.2.1.6 with the following information.

Such Service Level II Coatings used inside containment are procured to the same standards as Service Level I coatings with regard to radiation tolerance and performance under design basis accident conditions as discussed below.

Replace the second sentence of the third paragraph under the subsection titled "Service Level II Coatings" within DCD Subsection 6.1.2.1.6 with the following information.

Coating system application, inspection and monitoring requirements for the Service Level II coatings used inside containment will be performed in accordance with a program based on ASTM D5144 (Reference 201), "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants," and the guidance of ASTM D5163 (Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality requirements.

6.1.3 COMBINED LICENSE INFORMATION ITEMS

6.1.3.1 Procedure Review

STD COL 6.1-1 This COL Item is addressed in Subsection 6.1.1.2.

6.1.3.2 Coating Program

STD COL 6.1-2 This COL Item is addressed in Subsection 6.1.2.1.6.

6.1.4 REFERENCES

Add the following information at the end of DCD Subsection 6.1.4:

201. ASTM D5144-08, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants."

202. ASTM D5163-05a, "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant."

203. ASTM D7167-05, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant."

6.2 CONTAINMENT SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.2.4.5.1 Preoperational Inspection and Testing

LNP DEP 6.2-1

Hydrogen Ignition Subsystem

Revise the second paragraph of DCD Subsection 6.2.4.5.1, Hydrogen Ignition Subsystem, to read as follows:

Pre-operational inspection is performed to verify the location of openings through the ceilings of the passive core cooling system valve/accumulator rooms with respect to the containment pressure boundary. An analysis will be used to demonstrate that postulated hydrogen releases through these openings do not result in a failure of the containment shell.

6.2.5.1 Design Basis

Add the following information at the end of DCD Subsection 6.2.5.1, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, "Containment Leak Rate Test Program Description."

STD COL 6.2-1

The Containment Leak Rate Test Program using 10 CFR Part 50, Appendix J Option B is established in accordance with NEI 94-01 (DCD Subsection 6.2.7, Reference 30), as modified and endorsed by the NRC in Regulatory Guide 1.163. Table 13.4-201 provides milestones for containment leak rate testing implementation.

6.2.5.2.2 System Operation

Add the following information at the end of the subsection "Scheduling and Reporting of Periodic Tests" within DCD Subsection 6.2.5.2.2, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, "Containment Leak Rate Test Program Description."

STD COL 6.2-1

Schedules for the performance of periodic Type A, B, and C leak rate tests are in accordance with NEI 94-01, as endorsed and modified by Regulatory Guide 1.163, and described below:

<u>Type A Tests</u>

A preoperational Type A test is conducted prior to initial fuel load. If initial fuel load is delayed longer than 36 months after completion of the preoperational Type A test, a second preoperational Type A test shall be performed prior to initial fuel load. The first periodic Type A test is performed within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests are performed at a frequency of at least once per 48 months, until acceptable performance is established. The interval for testing begins at initial reactor operation. Each test interval begins upon completion of a Type A test and ends at the start of the next test. The extension of the Type A test interval is determined in accordance with NEI 94-01.

Type A testing is performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as successful completion of two consecutive Type A tests where the calculated performance leakage rate was less than 1.0 L_a. A preoperational Type A test may be used as one of the two Type A tests that must be successfully completed to extend the test interval, provided that an engineering analysis is performed to document why a preoperational Type A test can be treated as a periodic test. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

<u>Type B Tests (Except Containment Airlocks)</u>

Type B tests are performed prior to initial entry into Mode 4. Subsequent periodic Type B tests are performed at a frequency of at least once per 30 months, until acceptable performance is established. The test intervals for Type B penetrations may be increased based upon completion of two consecutive periodic as-found Type B tests where results of each test are within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the component prior to implementing Option B of 10 CFR Part 50, Appendix J. An extended test interval for Type B tests may be increased to a specific value in a range of frequencies from greater than once per 30 months up to a maximum of once per 120 months. The extension of specific test intervals for Type B penetrations is determined in accordance with NEI 94-01.

<u>Type B Tests (Containment Airlocks)</u>

Containment airlock(s) are tested at an internal pressure of not less than P_{ac}. (Prior to a preoperational Type A test $P_{ac} = P_a$.) Subsequent periodic tests are performed at a frequency of at least once per 30 months. In addition, equalizing valves, door seals, and penetrations with resilient seals (i.e., shaft seals, electrical penetrations, view port seals and other similar penetrations) that are testable, are tested at a frequency of once per 30 months.

For periods of multiple containment entries where the airlock doors are routinely used for access more frequently than once every seven days (e.g., shift or daily inspection tours of the containment), door seals may be tested once per 30 days during this time period.

Airlock door seals are tested prior to a preoperational Type A test. When containment integrity is required, airlock door seals are tested within seven days after each containment access.

Type C Tests

Type C tests are performed prior to initial entry into Mode 4. Subsequent periodic Type C tests are performed at a frequency of at least once per 30 months, until adequate performance has been established. Test intervals for Type C valves may be increased based upon completion of two consecutive periodic as-found Type C tests where the result of each test is within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the valve prior to implementing Option B of 10 CFR Part 50, Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 30 months up to a maximum of 60 months. Test interval extensions for Type C valves are determined in accordance with NEI 94-01.

Reporting

A post-outage report is prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The report is available on-site for NRC review. The report shows that the applicable performance criteria are met, and serves as a record that continuing performance is acceptable.

Add the following subsection at the end of DCD Subsection 6.2.5.2.2, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, “Containment Leak Rate Test Program Description.”

STD COL 6.2-1 Acceptance Criteria

Acceptance criteria for Type A, B and C Tests are established in Technical Specification 5.5.8.

6.2.6 COMBINED LICENSE INFORMATION FOR CONTAINMENT LEAK RATE TESTING

STD COL 6.2-1 This COL item is addressed in Subsections 6.2.5.1 and 6.2.5.2.2.

6.3 PASSIVE CORE COOLING SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.3.1.1.1 Emergency Core Decay Heat Removal

LNP DEP 3.2-1

Revise the first bullet and add new second and third bullets in the first paragraph of DCD Subsection 6.3.1.1.1 to read as follows:

- The passive residual heat removal heat exchanger automatically actuates to provide reactor coolant system cooling.
- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to Subsection 6.3.1.1.4).
- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for at least 72 hours following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in Subsection 6.3.1.1.4.

Replace the fourth bullet (old second bullet) in the first paragraph of DCD Subsection 6.3.1.1.1 with the following:

- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

LNP DEP 3.2-1
LNP DEP 6.3-1

Deleted - new fifth bullet (old third bullet)

6.3.1.1.4 Safe Shutdown

LNP DEP 3.2-1

Replace the first two paragraphs of DCD Subsection 6.3.1.1.4 with the following three paragraphs, to read:

The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions in the reactor coolant system.

The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

In most sequences the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to maintain safe, stable shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

Replace the first sentence of the fourth paragraph (old third paragraph) of DCD Subsection 6.3.1.1.4 with the following:

For loss of coolant accidents, when the core makeup tank level reaches the automatic depressurization system actuation setpoint and other postulated events where ac power sources are lost but passive residual heat removal heat exchanger operation is not extended or is exhausted, the automatic depressurization system will be initiated.

Replace the first sentence of the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4 with the following:

The basis used to define the passive core cooling system functional requirements is derived from Section 7.4 of the Standard Review Plan.

Add a last sentence to the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4, to read as follows:

The performance of the passive residual heat removal heat exchanger to bring the plant to 420°F in 36 hours is summarized in Subsection 19E.4.10.2.

6.3.1.1.6 Reliability Requirements

LNP DEP 3.2-1

Replace the last sentence of DCD Subsection 6.3.1.1.6 with the following:

Subsection 6.3.1.3 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

Add the following subsection after DCD Subsection 6.3.1.1.6.

DCD Subsection 6.3.1.2 is renumbered as Subsection 6.3.1.3.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Long-Term Core Decay Heat Removal

LNP DEP 3.2-1
LNP DEP 6.3-1

The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420°F in 36 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation. This non-bounding, conservative evaluation is discussed in subsection 19E.4.10.2.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for at least 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in Subsection 7.4.1.1.

LNP DEP 3.2-1
LNP DEP 6.3-1

6.3.1.3 Power Generation Design Basis

6.3.2.1 Schematic Piping and Instrumentation Diagrams

LNP DEP 3.2-1

Replace the first sentence of the first paragraph of DCD Subsection 6.3.2.1 with the following:

Figure 6.3-1 shows the piping and instrumentation drawings of the passive core cooling system.

6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

LNP DEP 3.2-1
LNP DEP 6.3-1

Replace the seventh and eighth paragraphs of DCD Subsection 6.3.2.1.1 with the following:

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and the passive containment cooling system, can provide core cooling for greater than 14 days. After the in-containment refueling water storage tank water reaches its saturation temperature (in several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter and downspouts normally drain to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for greater than 14 days.

LNP DEP 3.2-1

Revise the first and second sentences of the ninth paragraph of DCD Subsection 6.3.2.1.1 to read as follows:

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment.

Add a new tenth paragraph to DCD Subsection 6.3.2.1.1 to read as follows:

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Offsite or onsite ac power sources are typically recovered within a day, which would allow the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If ac power is not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

6.3.2.2.7 IRWST and Containment Recirculation Screens

LNP DEP 3.2-1

Replace the first paragraph of DCD Subsection 6.3.2.2.7 with the following:

The passive core cooling system has two different sets of screens that are used to prevent debris from entering the reactor and blocking core cooling passages during a LOCA: IRWST screens and containment recirculation screens. The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural frames attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including:

- GDC 35 of 10 CFR 50 Appendix A
- Regulatory Guide 1.82
- NUREG-0897

6.3.2.2.7.1 General Screen Design Criteria

LNP DEP 3.2-1

Add a new first paragraph to DCD Subsection 6.3.2.2.7.1 to read as follows:

The IRWST screens and containment recirculation screens are designed to comply with the following criteria.

6.3.2.2.7.2 IRWST Screens

LNP DEP 3.2-1

Replace the third paragraph of DCD Subsection 6.3.2.2.7.2 with the following:

During a LOCA, steam vented from the reactor coolant system condenses on the containment shell and drains down the shell to the polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell and is collected in a gutter near the operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.

6.3.2.8 Manual Actions

LNP DEP 3.2-1

Add a new third paragraph of DCD Subsection 6.3.2.8 to read as follows:

The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Add a new first sentence to the fourth paragraph (old third paragraph) of DCD Subsection 6.3.2.8, to read as follows:

Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation.

6.3.3 PERFORMANCE EVALUATION

LNP DEP 3.2-1

Replace the seventh paragraph of DCD Subsection 6.3.3 with the following:

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in Subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

Add a new eighth paragraph to DCD Subsection 6.3.3, as follows:

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition of 420°F within 36 hours as described in Subsection 19E.4.10.2. A non-bounding, conservative analysis of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of Subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

Add the following as the last sentence to the tenth paragraph (old ninth paragraph) of DCD Subsection 6.3.3, as follows:

If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in Subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

Add a new eleventh paragraph to DCD Subsection 6.3.3, as follows:

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in Subsection 6.3.1.1.4, the passive systems are capable of bringing the plant to a safe shutdown condition and maintaining that condition.

Add Subsection 6.3.3.2.1.1 to the end of DCD Subsection 6.3.3.2.1.

6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

LNP DEP 3.2-1

The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-loss of coolant accident event lasting as long as 24 hours, the automatic depressurization system will actuate if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat removal heat exchanger operation as described in Subsection 7.4.1.1, such that the automatic depressurization system does not actuate.

The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in Subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups

LNP DEP 6.3-1

Revise the last sentence of the fourth paragraph of DCD Subsection 6.3.3.4.1 to read as follows:

This allows it to function as a heat sink for greater than 14 days.

6.3.8 COMBINED LICENSE INFORMATION

6.3.8.1 Containment Cleanliness Program

Insert the following information at the end of DCD Subsection 6.3.8.1:

This COL Item is addressed below.

STD COL 6.3-1

Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris left in containment following personnel entry and exits. The program is consistent with the containment cleanliness program limits discussed in DCD Subsection 6.3.8.1. The program includes, as a minimum, the following:

Responsibilities

The program defines the organizational responsibilities for implementing the program; defines personnel and material controls; and defines the inspection and reporting requirements.

Implementation

Containment Entry/Exit

- Controls to account for the quantities and types of materials introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, to ensure adequate accountability controls. This may be accomplished by the work management process. Storage of aluminum is prohibited without engineering authorization. Cardboard boxes or miscellaneous packing material is not brought into containment without approval.
- If entries are made at power, prohibited materials and limits on quantities of materials that may generate hydrogen are established.

- Controls for loose items, such as keys and pens, which could be inadvertently left in containment.
- Methods and controls for securing any items and materials left unattended in containment.
- Administrative controls for accounting for tools, equipment and other material are established.
- Administrative controls for accounting of the permanent removal of materials previously introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, that may be left unattended in containment during outages and power operation. Types of materials considered are tape, labels, plastic film, and paper and cloth products.
- Requirements and actions to be taken for unaccounted for material.
- Requirements for final containment cleanliness inspections consistent with the design bases provided in DCD Subsection 6.3.8.1.
- Record keeping requirements for entry/exit logs.

Housekeeping

Housekeeping procedures require that work areas be maintained in a clean and orderly fashion during work activities and returned to original conditions (or better) upon completion of work.

Sampling Program

A sampling program is implemented consistent with NEI Guidance Report 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" as supplemented by the NRC in the "Safety Evaluation by The Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), 'Pressurized Water Reactor Sump Performance Evaluation Methodology.'" Latent debris sampling is implemented before startup. The sampling is conducted after containment exit cleanliness inspections to provide reasonable assurance that the plant latent debris design bases are met. Sampling frequency and scope may be adjusted based on sampling results. Results are evaluated post-start up and any nonconforming results will be addressed in the Corrective Action Program.

6.4 HABITABILITY SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

LNP DEP 6.4-1

Revise the first sentence of the third paragraph of DCD Section 6.4 to read as follows:

If ac power is unavailable for more than 10 minutes or if "High-2" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES).

6.4.2.2 General Description

LNP DEP 6.4-2

Revise the sixth paragraph of DCD Subsection 6.4.2.2 to read as follows:

In the unlikely event that power to the nuclear island nonradioactive ventilation system is unavailable for more than 72 hours, MCR habitability is maintained by operating one of the two MCR ancillary fans to supply outside air at or below the maximum normal site ambient temperature to the MCR. See subsection 9.4.1 for a description of this cooling mode of operation. Doors and ducts may be opened to provide a supply pathway and an exhaust pathway. Likewise, outside air is supplied to division B and C instrumentation and control rooms in order to maintain the ambient temperature below the qualification temperature of the equipment.

6.4.2.3 Component Description

LNP DEP 6.4-2

Revise the first paragraph of DCD Subsection 6.4.2.3 and add a new fourteenth bullet to read as follows:

The main control room emergency habitability system compressed air supply contains a set of storage tanks connected to a main and an alternate air delivery line and equipment to provide electrical load de-energization. Components common to both lines include a manual isolation valve and a pressure regulating valve. Single active failure protection is provided by the use of redundant, remotely operated isolation valves, which are located within the MCR pressure boundary. In the event of insufficient or excessive flow in the main delivery line, the main delivery line is isolated and the alternate delivery line is manually actuated. The alternate delivery line contains the same components as the main delivery line with the exception of the remotely operated isolation valves, and thus is capable of supplying compressed air to the MCR pressure boundary at the required air flowrate. The VES piping and penetrations for the MCR envelope are designated as equipment Class C. Additional details on Class C designation are provided in Subsection 3.2.2.5. The classification of VES components is provided in Table 3.2-3, as appropriate.

- MCR Load Shed Panels

The de-energization of the Main Control Room (MCR) electrical loads will be performed using Class 1E equipment. Equipment within each of the two electrical panels will be actuated from the "main control room isolation, air supply initiation and electrical load de-energization" engineered safety feature. The de-energization is separated into two stages to provide operators with the maximum available non-safety equipment, while maintaining the MCR heat load within the requirements of the VES.

Each electrical panel will have redundant relays and timers controlled by both PMS Division A and PMS Division C. Either division will be capable of actuating the timers and relays associated with each electrical panel independent of one another. This configuration prevents routine maintenance or single failures of a PMS cabinet from creating a spurious loss of MCR electrical loads, while still providing for single failure protection. In order to accomplish the "De-energize MCR Electrical Loads" function, one set of Stage 1 and Stage 2 timers in each electrical panel must receive the PMS command.

Relays in both electrical panels must be actuated in order to carry out the overall function; however, overall actuation may occur via different combinations of Division A and Division C commands.

6.4.2.6 Shielding Design

LNP DEP 6.4-1

Revise DCD Subsection 6.4.2.6 to read as follows:

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the main control room. Main control room shielding design bases are discussed in Section 12.3. In addition to shielding provided by building structural features, consideration is given to shielding provided by the VES filter shielding. Descriptions of the design basis LOCA source terms, main control room shielding parameters, and evaluation of doses to main control room personnel are presented in Section 15.6.

The main control room and its location in the plant are shown in Figure 12.3-1.

6.4.3 SYSTEM OPERATION

Add the following information at the end of DCD Subsection 6.4.3:

STD COL 6.4-2

Generic Issue 83 addresses the importance of maintaining control room habitability following an accidental release of external toxic or radioactive material or smoke and the capability of the control room operators to safely control the reactor. Procedures and training for control room habitability are written in accordance with Section 13.5 for control room operating procedures, and Section 13.2 for operator training. The procedures and training are verified to be consistent to the intent of Generic Issue 83.

The procedures and training address the toxic chemical events addressed in Sections 2.2 and 6.4 consistent with the guidance provided in regulatory position C.5 of Regulatory Guide 1.78, including arrangements with Federal, State, and local agencies or other cognizant organizations for the prompt notification of the nuclear power plant when accidents involving hazardous chemicals occur within five miles of the plant. The procedures include the conduct of periodic surveys of stationary and mobile sources of hazardous chemicals affecting the evaluations consistent with the guidance provided in regulatory position 2.5 of Regulatory Guide 1.196. The procedures include appropriate reviews of the configuration of the control room envelope and habitability systems consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196. The procedures also include periodic assessments of the control room habitability systems' material condition, configuration controls, safety analyses, and operating and maintenance procedures consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196.

Procedures for testing and maintenance are consistent with the design requirements of the DCD including the guidance provided in regulatory position 2.7.1 of Regulatory Guide 1.196.

6.4.3.2 Emergency Mode

LNP DEP 6.4-1

Revise the first bullet of the first paragraph of DCD Subsection 6.4.3.2 to read as follows:

- "High-2" particulate or iodine radioactivity in the main control room supply air duct

Revise the first sentence of the second paragraph of DCD Subsection 6.4.3.2 to read as follows:

The nuclear island nonradioactive ventilation system is isolated from the main control room pressure boundary by automatic closure of the isolation devices located in the nuclear island nonradioactive ventilation system ductwork if radiation levels in the main control room supply air duct exceed the "High-2" setpoint or if ac power is lost for more than 10 minutes.

LNP DEP 6.4-2

Revise the fifth paragraph of DCD Subsection 6.4.3.2 to read as follows:

The temperature and humidity in the main control room pressure boundary following a loss of the nuclear island nonradioactive ventilation system remain within limits for reliable human performance (References 2 and 3) over a 72-hour period. The temperature/relative humidity values calculated during the 72 hours following a design basis accident equate to an effective temperature of 85°F or less (considering operators performing light work, with no credit for the cooling effect of air velocity, and a relative humidity of less than 50 percent) to support reliable human performance constraints (References 2 and 3). Non-1E MCR heat loads are de-energized by PMS automatic actions and the 24 hour battery heat loads are terminated or exhausted at 24 hours to maintain the assumed heat load values in Table 6.4-3, which then maintains the occupied zone of the MCR and the zones containing qualified safety-related equipment within the temperature constraints at 72-hours post VES actuation. The occupied zone is considered to be the area between the raised floor and 7 ft. above the floor which encompasses the reactor operator and senior reactor operator consoles.

6.4.4 SYSTEM SAFETY EVALUATION

LNP DEP 6.4-1

Revise the third paragraph of DCD Subsection 6.4.4 to read as follows:

Doses were determined for the following design basis:

	VES Operating	VBS Operating
Large Break LOCA	4.33 rem TEDE	4.84 rem TEDE
Fuel Handling Accident	1.5 rem TEDE	1.6 rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	3.4 rem TEDE	4.0 rem TEDE
(Accident-initiated iodine spike)	1.0 rem TEDE	1.4 rem TEDE
Steam Line Break		
(Pre-existing iodine spike)	1.1 rem TEDE	0.9 rem TEDE
(Accident-initiated iodine spike)	1.3 rem TEDE	2.9 rem TEDE
Rod Ejection Accident	3.6 rem TEDE	2.5 rem TEDE
Locked Rotor Accident		
(Accident without feedwater available)	0.4 rem TEDE	0.7 rem TEDE
(Accident with feedwater available)	0.2 rem TEDE	0.9 rem TEDE
Small Line Break Outside Containment	0.4 rem TEDE	0.4 rem TEDE

Add the following information after the second sentence of the eighth paragraph of DCD Subsection 6.4.4.

LNP SUP 6.4-1

The site does not plan to use any gas dispersants. The use of liquid dispersants is discussed in Section 10.4.

Insert the following information at the end of the eighth paragraph of DCD Subsection 6.4.4.

STD COL 6.4-1

Table 6.4-201 provides additional details regarding the evaluated onsite chemicals.

LNP DEP 6.4-1

Revise the first bullet of the thirteenth paragraph of DCD Subsection 6.4.4 to read as follows:

- “High-2” particulate or iodine radioactivity in MCR air supply duct

Revise the last sentence of the sixteenth paragraph of DCD Subsection 6.4.4 to read as follows:

The following cases are evaluated since they involve releases that extend beyond 24 hours after the initiation of the event:

Large Break LOCA	4.4 rem TEDE
Steam Line Break	
(Pre-existing iodine spike)	1.2 rem TEDE
(Accident-initiated iodine spike)	2.0 rem TEDE

Insert the following subsections at the end of DCD Subsection 6.4.4.

6.4.4.1 Dual Unit Analysis

STD SUP 6.4-1

Credible events that could put the control room operators at risk from a dose standpoint at a single AP1000 unit have been evaluated and addressed in the DCD. The dose to the control room operators at an adjacent AP1000 unit due to a radiological release from another unit is bounded by the dose to control room operators on the affected unit. While it is possible that a unit may be downwind in an unfavorable location, the dose at the downwind unit would be bounded by what has already been evaluated for a single unit AP1000. Simultaneous accidents at multiple units at a common site are not considered to be a credible event.

6.4.4.2 Toxic Chemical Habitability Analysis

LNP COL 6.4-1

Subsection 2.2.3 determined that there are no design basis events due to site-specific sources of hazardous materials in the vicinity of the plant that require mitigating actions to be undertaken to eliminate or lessen the likelihood and severity of potential accidents.

6.4.5.1 Preoperational Inspection and Testing

LNP DEP 6.4-2

Revise the third paragraph of DCD Subsection 6.4.5.1 to read as follows:

Temperatures within the MCR where the operators are located are verified by analysis and/or testing to remain within limits for reliable human performance (References 2 and 3) for a 72-hour period following a bounding scenario with MCR isolation and non-safety ac power available (see Table 6.4-3 for heat loads assumed in analysis) and a station blackout (battery backed loads only).

6.4.7 COMBINED LICENSE INFORMATION

LNP COL 6.4-1
STD COL 6.4-1

This COL Item is addressed in Subsections 6.4.4 and 6.4.4.2.

STD COL 6.4-2

This COL Item is addressed in Subsection 6.4.3.

Table 6.4-201 (Sheet 1 of 2)
MAIN CONTROL ROOM HABITABILITY EVALUATIONS OF ONSITE TOXIC CHEMICALS[(1)]

STD COL 6.4-1

A – STANDARD ONSITE TOXIC CHEMICALS

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
Hydrogen	Gas	500 scf	126.3 ft	Yard at turbine building	MCR
Hydrogen	Liquid	1500 gal	577 ft	Gas storage	MCR
Nitrogen	Liquid	3000 gal	577 ft	Gas storage	MCR
Carbon Dioxide (CO_2)	Liquid	6 tons	577 ft	Gas storage	MCR
Oxygen Scavenger [Hydrazine]	Liquid	1600 gal	203 ft	Turbine building	IH
pH Addition [Morpholine]	Liquid	1600 gal	203 ft	Turbine building	IH
Sulfuric Acid	Liquid	800 gal	203 ft	Turbine building	IH
Sulfuric Acid	Liquid	20,000 gal	436 ft	CWS area	IH
Sodium Hydroxide	Liquid	800 gal	203 ft	Turbine building	S
Sodium Hydroxide	Liquid	20,000 gal	436 ft	CWS area	S
Fuel Oil	Liquid	60,000 gal	197 ft	DG fuel oil storage tank, DG building, Annex building	IH
Corrosion Inhibitor [Sodium Molybdate]	Liquid	800 gal	203 ft	Turbine building	S
Corrosion Inhibitor [Sodium Molybdate]	Liquid	10,000 gal	436 ft	CWS area	S
Scale Inhibitor [Sodium Hexametaphosphate]	Liquid	800 gal	203 ft	Turbine building	S
Scale Inhibitor [Sodium Hexametaphosphate]	Liquid	10,000 gal	436 ft	CWS area	S
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	800 gal	203 ft	Turbine building	S
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	10,000 gal	436 ft	CWS area	S
Algaecide [Ammonium comp. polyethoxylate]	Liquid	800 gal	203 ft	Turbine building	S
Algaecide [Ammonium comp. polyethoxylate]	Liquid	10,000 gal	436 ft	CWS area	S

Table 6.4-201 (Sheet 2 of 2)
MAIN CONTROL ROOM HABITABILITY EVALUATIONS OF ONSITE TOXIC CHEMICALS[(1)]

LNP COL 6.4-1

B – SITE SPECIFIC ONSITE TOXIC CHEMICALS

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
None currently identified					

Notes:

STD COL 6.4-1 1) This table supplements DCD Table 6.4-1. Quantities are by largest evaluated container content for the evaluated location per unit. Quantities and distances are bounding evaluation values and may not be actual amounts and distances. Smaller quantities of a chemical at further distances from the MCR air intake are not shown on this table. Actual site locations are confirmed to be at or beyond the evaluated distance.

S – Chemicals with an Impact Evaluation designation of “S” for the MCR Habitability Impact Evaluation were evaluated and screened out based on the chemical properties, distance, and quantities.

IH – Chemicals with an Impact Evaluation designation of “IH” indicates the evaluation of this chemical considered the design detail of the main control room intake height.

MCR – Chemicals with an Impact Evaluation designation of “MCR” indicates the evaluation of this chemical considered design details of the main control room such as volume, envelope boundaries, ventilation systems, and occupancy factor.

Table 6.4-202
Main Control Room Habitability Indications and Alarms

LNP DEP 6.4-1

VES emergency air storage tank pressure (indication and low and low-low alarms)

VES MCR pressure boundary differential pressure (indication and high and low alarms)

VES air delivery line flowrate (indication and high and low alarms)

VES passive filtration flow rate (indication and high and low alarms)

VBS main control room supply air radiation level (High-1 and-High-2 alarms)

VBS outside air intake smoke level (high alarm)

VBS isolation valve position

VBS MCR pressure boundary differential pressure

Table 6.4-203
Loss of AC Power Heat Load Limits

LNP DEP 6.4-2

Room Name	Room Numbers	Heat Load 0 to 24 Hours (Btu/sec)	Heat Load 24 to 72 Hours (Btu/sec)
MCR Envelope	12401	26.1 (Hour 0 through 0.5) 15.6 (Hour 0.5 through 3.5) 5.8 (Hour 3.5 through 24)	2.9
I&C Rooms	12301, 12305	8.8	0
I&C Rooms	12302, 12304	13.0	4.2
dc Equipment Rooms	12201, 12205	3.7 (Hour 0 through 1) 2.4 (Hour 2 through 24)	0
dc Equipment Rooms	12203, 12207	5.8 (Hour 0 through 1) 4.5 (Hour 2 through 24)	2.0

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

6.6 INSERVICE INSPECTION OF CLASS 2, 3, AND MC COMPONENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following to DCD Section 6.6 ahead of Subsection 6.6.1 heading:

STD COL 6.6-1

The initial inservice inspection program incorporates the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load. Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in Regulatory Guide 1.147, that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed in 10 CFR 50.55a(b)).

6.6.1 COMPONENTS SUBJECT TO EXAMINATION

Add the following to the end of DCD Subsection 6.6.1:

STD COL 6.6-1

Class 2 and 3 components are included in the equipment designation list and the line designation list contained in the inservice inspection program.

6.6.2 ACCESSIBILITY

Revise the first and last sentences of the third paragraph in DCD Subsection 6.6.2 to add supplemental information as follows:

STD SUP 6.6-1

Considerable experience has been drawn on in designing, locating, and supporting Quality Group B and C (ASME Class 2 and 3) and Class MC pressure-retaining components to permit pre-service and inservice inspection required by Section XI of the ASME Code. Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selections are used in establishing the designs. The inspection design goals are to eliminate uninspectable components, reduce occupational radiation exposure, reduce inspection times, allow state-of-the-art inspection systems, and enhance detection and the reliability of flaw characterization. There are no Quality Group B and C components or Class MC components, which require inservice inspection during reactor operation.

Add the following to the end of DCD Subsection 6.6.2:

STD COL 6.6-2

During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Modifications reviewed following design certification adhere to the same level of review as the certified design per 10 CFR Part 50, Appendix B as implemented by the Westinghouse Quality Management System (QMS). The QMS requires that changes to approved design documents, including field changes, are subject to the same review and approval process as the original design. This explicitly requires the field change process to follow the same level of review that was required during the design process. Accessibility and inspectability are key components of the design process.

Control of accessibility for inspectability and testing during post-design certification activities is provided via procedures for design control and plant modifications.

6.6.3 EXAMINATION TECHNIQUES AND PROCEDURES

Add the following Subsections 6.6.3.1, 6.6.3.2 and 6.6.3.3 to the end of DCD Subsection 6.6.3:

6.6.3.1 Examination Methods

Visual Examination

STD COL 6.6-1

Visual examination methods VT-1, VT-2 and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided in accordance with Table IWA-2210-1.

Surface Examination

Magnetic particle, liquid penetrant, and eddy current examination techniques are performed in accordance with ASME Section XI, IWA-2221, IWA-2222, and IWA-2223 respectively. Direct examination access for magnetic particle (MT) and liquid penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (see Visual Examination), except that additional access is provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision.

Ultrasonic Examination

Volumetric ultrasonic direct examination is performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I.

Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. In accordance with 10 CFR 50.55a(b)(2)(xix), IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.

6.6.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII.

6.6.3.3 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 10 CFR 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

6.6.4 INSPECTION INTERVALS

Add the following to the end of DCD Subsection 6.6.4:

STD COL 6.6-1 Because 10 CFR 50.55a(g)(4) requires 120-month inspection intervals, Inspection Program B of IWB-2400 must be chosen. The inspection interval is divided into three periods. Period one comprises the first three years of the interval, period two comprises the next four years of the interval, and period three comprises the remaining three years of the inspection interval. The periods within each inspection interval may be extended by as much as one year to permit inspections to be concurrent with plant outages. The adjustment of period end dates shall not alter the rules and requirements for scheduling inspection intervals. It is intended that inservice examinations be performed during normal plant outages, such as refueling shutdown or maintenance shutdowns occurring during the inspection interval.

6.6.6 EVALUATION OF EXAMINATION RESULTS

Add the following new paragraph at the end of DCD Subsection 6.6.6:

STD COL 6.6-1

Components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWC-3122.3 or IWC-3132.3 for Class 2 components, IWD-3000 for Class 3 components, IWE-3122.3 for Class MC components, or IWF-3112.2 or IWF-3122.2 for component supports, are subjected to successive period examinations in accordance with the requirements of IWC-2420, IWD-2420, IWE-2420, or IWF-2420, respectively. Examinations that reveal flaws or relevant conditions exceeding Table IWC-3410-1, IWD-3000, IWE-3000, or IWF-3400 acceptance standards are extended to include additional examinations in accordance with the requirements of IWC-2430, IWD-2430, or IWF-2430, respectively.

6.6.9 COMBINED LICENSE INFORMATION ITEMS

6.6.9.1 Inspection Programs

STD COL 6.6-1

This COL Item is addressed in Section 6.6 introduction, and in Subsections 6.6.1, 6.6.3.1, 6.6.3.2, 6.6.3.3, 6.6.4, and 6.6.6.

6.6.9.2 Construction Activities

STD COL 6.6-2

This COL Item is addressed in Subsection 6.6.2.

APPENDIX 6A
FISSION PRODUCT DISTRIBUTION IN THE AP1000 POST-DESIGN BASIS ACCIDENT CONTAINMENT ATMOSPHERE

This section of the referenced DCD is incorporated by reference with no departures or supplements.