APR1400
DESIGN CONTROL DOCUMENT TIER 2

CHAPTER 3
DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS, AND EQUIPMENT

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<td>air cleaning unit</td>
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<td>heating, ventilation, and air conditioning</td>
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<td>Joint Owner Group</td>
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<td>LBB</td>
<td>leak before break</td>
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| LC           | 1) lock close  
<p>|              | 2) loop controller |
| LCO          | limiting conditions for operation |</p>
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<tr>
<td>SAM</td>
<td>seismic anchor movement motion</td>
</tr>
</tbody>
</table>
| SC           | 1) shutdown cooling  
<pre><code>                                    | 2) safety console                  |
</code></pre>
<p>| SCP          | shutdown cooling pump               |
| SCS          | shutdown cooling system             |
| SDCHX        | shutdown cooling heat exchanger     |
| SECY         | office of the secretary of the commission |
| SFP          | spent fuel pool                     |
| SG           | steam generator                     |
| SGBS         | steam generator blowdown system     |</p>
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>SGTR</td>
<td>steam generator tube rupture</td>
</tr>
<tr>
<td>SI</td>
<td>safety injection</td>
</tr>
<tr>
<td>SIF</td>
<td>stress intensification factor</td>
</tr>
<tr>
<td>SIP</td>
<td>safety injection pump</td>
</tr>
<tr>
<td>SIS</td>
<td>safety injection system</td>
</tr>
</tbody>
</table>
| SIT     | 1) safety injection tank  
<pre><code>        | 2) structural integrity test |
</code></pre>
<p>| SLB     | steam line break |
| SMAW    | shielded metal arc weld |
| SOV     | solenoid-operated valve |
| SQSDS   | seismic qualification summary data sheet |
| SRI     | Stanford Research Institute |
| SRP     | Standard Review Plan |
| SRSS    | square root of the sum of the squares |
| SRV     | safety relief valve |
| SS      | outside secondary shield |
| SSC     | structures, systems, and components |
| SSE     | safe shutdown earthquake |
| SSI     | soil structure interaction |
| SSW     | secondary shield wall |
| T/C     | reactor inlet temperature, ( T(\text{cold}) ) |
| T/H     | reactor outlet temperature, ( T(\text{hot}) ) |
| TEDE    | total effective dose equivalent |
| TGB     | turbine generator building |
| TIV     | temperature isolation valve |
| TLOFW   | total loss of feedwater |
| UGS     | upper guide structure |
| UHS     | ultimate heat sink |
| UPC     | ultimate pressure capacity |
| URS     | uniform response spectrum |
| USH     | uniform support motion |</p>
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>VCT</td>
<td>volume control tank</td>
</tr>
<tr>
<td>WR</td>
<td>wide range</td>
</tr>
<tr>
<td>ZPA</td>
<td>zero period acceleration</td>
</tr>
</tbody>
</table>
CHAPTER 3 – DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS, AND EQUIPMENT

3.1 Conformance with Nuclear Regulatory Commission General Design Criteria

3.1.1 Criterion 1 – Quality Standards and Records

“Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.”

Response:

The structures, systems, and components (SSCs) described in the Design Control Document (DCD) are classified according to their importance to safety, including the prevention and mitigation of accidents using the classification system described in ANSI/ANS 51.1 (Reference 1). Each component is given a safety class designation. The codes, standards, and quality control applicable to each component and safety class designation are identified in Section 3.2. The design and fabrication of SSCs conform with 10 CFR 50.55a (Reference 2) as applicable. The quality assurance program conforms with the requirements of 10 CFR Part 50, Appendix B (Reference 3), and is addressed in Section 17.5. Chapter 14 describes the initial tests and operations that are conducted on installed equipment to provide reasonable assurance that the performance of the equipment is commensurate with the importance of the safety function.

The design, fabrication, and quality programs for components not included in the American National Standards Institute (ANSI) classification system are governed by industry codes
appropriate to the application. Additional details are described in the relevant DCD sections.

3.1.2 Criterion 2 – Design Bases for Protection Against Natural Phenomena

“Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.”

Response:

The SSCs important to safety are designed to accommodate, without loss of capability, the effects of the design basis natural phenomena along with appropriate combinations of normal and accident conditions, as described in this chapter. Additional design information is provided in the sections that describe the individual SSCs.

3.1.3 Criterion 3 – Fire Protection

“Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.”

Response:

The pressure boundary components and structures and the attendant auxiliary systems in the APR1400 design scope are designed to minimize the probability and effects of fires and
explosions. High-grade noncombustible and fire-resistant materials are used for components located in the containment, components of engineered safety feature systems, and throughout the plant wherever practical. The fire protection system is described in Subsection 9.5.1.

3.1.4 Criterion 4 – Environmental and Dynamic Effects Design Bases

“Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.”

Response:

SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a LOCA as described in Section 3.11.

Where appropriate, design bases include design requirements that provide reasonable assurance that SSCs are appropriately protected against dynamic effects such as missiles, pipe whipping, fluid discharges that could result from equipment failures, postulated accidents, and events and conditions outside the nuclear power plant.

The design concept of leak-before-break (LBB) is applied to the reactor coolant piping including surge line, safety injection, and shutdown cooling piping inside containment. The LBB concept eliminates the dynamic effects of postulated pipe ruptures from the design basis and is carried out in accordance with SRP 3.6.3 (Reference 4) and NUREG-1061 (Reference 5), Volume 3. The applications of LBB provisions are described in Subsection 3.6.3.
3.1.5  **Criterion 5 – Sharing of Structures, Systems, and Components**

“Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.”

**Response:**

SSCs that perform safety-related functions are not shared between two units because the APR1400 is a single-unit plant.

3.1.6  **Criterion 10 – Reactor Design**

“The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.”

**Response:**

Specified acceptable fuel design limits (SAFDLs) are described in Subsection 4.4.1. Operation within the limiting conditions for operation (LCOs) specified by the Technical Specifications keeps the reactor fuel within the SAFDLs for normal operation, including anticipated operational occurrences (AOOs).

The plant is designed so that operation within the LCOs and limiting safety system settings (LSSSs) prescribed in the Technical Specifications results in confidence that SAFDLs are not exceeded as a result of an AOO. Operator action, aided by control systems and monitored by plant instrumentation, maintains the plant within LCO limits during normal operation.

Additional information is provided in the following sections:

a.  Section 4.2, Fuel System Design
b.  Chapter 5, Reactor Coolant System and Connecting Systems
c.  Subsection 5.4.7, Shutdown Cooling System
d. Section 7.2, Reactor Trip System

e. Chapter 15, Transient and Accident Analyses

f. Chapter 16, Technical Specifications

3.1.7 **Criterion 11 – Reactor Inherent Protection**

“The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.”

**Response:**

In the power operating range, the combined response of the fuel temperature coefficient (FTC), moderator temperature coefficient (MTC), moderator void coefficient, and moderator pressure coefficient to an increase in reactor power is a decrease in reactivity (i.e., the inherent nuclear feedback characteristics are not positive).

The reactivity coefficients for this reactor are described in Subsection 4.3.1.4.

3.1.8 **Criterion 12 – Suppression of Reactor Power Oscillations**

“The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.”

**Response:**

Total reactor power oscillations of the fundamental mode stability is provided by the negative power coefficient of reactivity (GDC 11, Subsection 3.1.7) and the coolant temperature program maintained by regulating rods. Power level is continuously monitored by neutron flux detectors (Chapter 7).

Power distribution oscillations are detected by neutron flux detectors. Axial mode oscillations are suppressed by means of part-strength or full-strength neutron absorber rods. All other modes of oscillation are expected to be convergent. Monitoring and protective
requirements imposed by GDC 10 and 20 are described in Subsections 3.1.6 and 3.1.16 and Chapter 4.

3.1.9 Criterion 13 – Instrumentation and Control

“Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.”

Response:

Instrumentation is provided to monitor significant process variables that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and their associated systems. Controls are provided to maintain these variables within the limits prescribed for safe operation. Instrumentation and control for the containment and its associated systems are described in Chapters 6 through 12. The principal process variables to be monitored and controlled are:

a. Neutron flux level (reactor power)

b. Control element assembly (CEA) positions

c. Neutron flux distribution (at various axial positions)

d. Reactor coolant temperature and pressure

e. Reactor coolant pump (RCP) speed

f. Pressurizer (PZR) level

g. Steam generator (SG) level and pressure

The departure from nucleate boiling ratio (DNBR) margin and local power density (LPD) margin (kW/cm) are also monitored.
The plant protection system (PPS) consists of the reactor protection system (RPS) and engineered safety features actuation system (ESFAS). The RPS is designed to monitor nuclear steam supply system (NSSS) operating conditions and initiate reliable and rapid reactor shutdown if monitored variables or combinations of monitored variables deviate from the permissible operating range to a degree that a safety limit may be reached. The ESFAS is designed to monitor plant variables and to actuate ESF systems during a design basis event (DBE).

The following systems and equipment are provided to monitor and maintain control over the fission process during transient and steady-state periods over the life of the core:

a. Redundant channels of ex-core nuclear instrumentation as the primary means of monitoring the fission process for protection, control, and low power operation

b. Redundant and diverse CEA position-indicating systems

c. Manual and automatic control of reactor power by means of CEAs

d. Manual regulation of boron concentration in the reactor coolant

e. A boronometer to determine boron concentration in the reactor coolant by neutron absorption as a backup to routine sampling and analysis

f. In-core instrumentation, provided to supplement information on core power distribution and enable calibration of ex-core flux detectors

The non-nuclear instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant, main steam, and auxiliary systems and is used to maintain these variables within the prescribed limits. The instrumentation and control (I&C) systems are described in Chapter 7. The boronometer is described in Subsections 7.7.1.1 and 9.3.4.2. The process radiation monitor is described in Subsection 9.3.4.2.

When a variable is to be monitored during and after a DBE, in addition to normal operation, the event analysis results are used to provide reasonable assurance that the instruments provided cover the range anticipated for the event conditions.
3.1.10 Criterion 14 – Reactor Coolant Pressure Boundary

“The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.”

Response:

The RCPB is defined in accordance with 10 CFR 50.2 and ANSI/ANS 51.1, as described in the response to GDC 55, Subsection 3.1.48.

Reactor coolant system (RCS) components are designed to meet the requirements of ASME Section III. To establish operating pressure and temperature limitations during startup and shutdown of the RCS, the fracture toughness rules defined in ASME Section III are followed. Quality control, inspection, and testing are performed as required by ASME Section III and allowable reactor pressure-temperature operations are specified to provide reasonable assurance of the integrity of the RCS.

The RCPB is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation, including anticipated transients, and maintain the stresses within applicable limits.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures that produce welds of complete fusion and free of unacceptable defects are followed. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of ASME Section IX for the materials to be welded. Qualification records, including the results of the procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

The pressure boundary has provisions for in-service inspection in accordance with ASME Section XI to provide reasonable assurance of the continuance of the structural and leak-tight integrity of the boundary, as described in the response to GDC 32, Subsection 3.1.28. For the reactor vessel, a material surveillance program conforming to the requirements of 10 CFR Part 50, Appendix H, is provided.
3.1.11 Criterion 15 – Reactor Coolant System Design

“The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.”

Response:

The design criteria and bases for the RCPB are described in the response to GDC 14.

The operating conditions for normal steady-state and transient plant operations are established conservatively. Normal operating limits are selected so that an adequate margin exists between normal operating limits and design limits. The plant control systems are designed to provide reasonable assurance that plant variables are maintained well within established operating limits. The plant transient response characteristics and pressure and temperature distributions during normal operations are considered in the design as well as the accuracy and response of the instruments and controls. These design techniques provide reasonable assurance that a satisfactory margin is maintained between normal operating conditions, including design transients, and design limits for the RCPB.

Plant control systems function to minimize the deviations from normal operating limits in the event of AOOs. When the control system is inadequate or fails to respond upon demand, the plant protection system starts functioning to mitigate the consequences of such events.

The plant protection system functions to mitigate the consequences in the event of accidents. Analyses show that design limits for the RCPB are not exceeded in the event of any ANSI/ANS 51.1 conditions.

3.1.12 Criterion 16 – Containment Design

“Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.”
Response:

The containment design incorporates a post-tensioned concrete containment with a steel liner to enclose the nuclear steam supply system completely. The containment system is designed to protect the public from the consequences of a LOCA, based on the equivalent energy release of a postulated break of reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe.

The containment building and the associated engineered safety feature systems are designed to safely withstand all internal and external environmental conditions that may reasonably be expected to occur during the life of the plant, including both short- and long-term effects of a LOCA.

The design criteria and methods of analysis for the containment structure are described in Subsections 3.8.1 and 3.8.2. The leak-tightness of the containment system and short- and long-term performance following a LOCA are described in Section 6.2.

3.1.13 Criterion 17 – Electrical Power Systems

“An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified
acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.”

Response:

The APR1400 design is provided with an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety in conformance with the requirements of this GDC, as described in Chapter 8.

The onsite electric power system consists of separate, redundant, and independent distribution systems and dedicated power supplies with sufficient capacity, capability, and testability to perform associated safety functions assuming a single failure.

The offsite electric power is supplied to the plant from the transmission network by at least two independent circuits through the site-specific switchyard. Each circuit is immediately available and has sufficient capacity and capability to perform the associated safety function.

Provisions are made to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit.

3.1.14 Criterion 18 – Inspection and Testing of Electrical Power Systems

“Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that
brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.”

Response:

Electrical power systems important to safety are designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and to detect deterioration, if any, of their components. Capability is provided to periodically test the operability and functional performance of the system components. The diesel generator sets are started and loaded periodically on a routine basis, and relays, switches, and buses are inspected and tested for operation and availability.

Transfer from normal to emergency sources of power is performed to check system operability and the full operational sequence that brings the systems into operation.

Further information is provided in Subsections 8.3.1 and 8.3.2 and in Chapter 16.

3.1.15 Criterion 19 – Control Room

“A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses
using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.”

Response:

All control and monitoring equipment necessary to operate or shut the unit down and maintain safe control of the facility is located in the main control room (MCR).

The design of the MCR permits safe occupancy during abnormal conditions. The use of non-combustible and fire-retardant materials in the construction of the MCR, the limitation of combustible supplies, the location of firefighting equipment, and the continuous presence of a trained operator minimize the possibility that the MCR will become uninhabitable. Radiation exposure levels following design basis accidents are maintained below allowable levels by proper design of shielding and ventilation. The MCR ventilation system is designed to allow access to occupancy of the MCR under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE during the accident. Fission product removal is performed by the MCR heating, ventilation and air conditioning (HVAC) system to remove iodine and particle matter, thereby minimizing the dose that could result from the accident. Radiation detectors and alarms are provided. MCR habitability features are described in Section 6.4. Emergency lighting is described in Subsection 9.5.3.

If the MCR is inaccessible, alternate local controls and instruments are available for equipment required to establish and maintain a hot standby condition, as well as cold shutdown. Hot and cold shutdown capability is addressed in Section 7.4 for the systems required for safe shutdown. The MCR is described in Subsection 7.7.1.2. Human factors issues are addressed in Chapter 18.

Radiation protection for the plant’s control facilities is addressed in Section 6.4 and Chapter 2.

3.1.16 Criterion 20 – Protection System Functions

“The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational
occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.”

Response:

A plant protection system (PPS), consisting of a reactor protection system (RPS) and an engineered safety features actuation system (ESFAS), is provided. The RPS automatically initiates a reactor trip when any of the monitored process variables reaches a trip setpoint. The ESFAS automatically actuates engineered safety feature (ESF) systems and their support systems when any of the monitored process variables reaches a predetermined setpoint.

The trip setpoints of the RPS are selected to provide reasonable assurance that DBEs that are expected to occur once or more during the life of the nuclear generating station do not cause violation of SAFDLs. The reactor trips also help the ESF systems in mitigating the consequences of DBEs that are expected to occur once during the life of several plants as well as arbitrary combinations of unplanned events and degraded systems that are never expected to occur, to within acceptable limits.

Reactor trip is accomplished by de-energizing the control element drive mechanism (CEDM) coils through the interruption of the CEDM power supply either automatically or manually. The CEDM power supply is a pair of full-capacity motor-generator sets. The path to the CEDMs is interrupted by opening the reactor trip switchgear. With the CEDM coils de-energized, the CEAs are released to drop into the core by gravity, rapidly inserting negative reactivity to shut the reactor down. The CEDMs are addressed in Subsection 3.9.4 and specific reactor trips are addressed in Section 7.2.

The ESF systems are actuated to minimize the effects of incidents that could occur. Controls are provided for manual actuation of the ESF system. The ESF systems are addressed in Chapter 6. The process variables that automatically actuate the ESF system and the logic for the ESFAS are addressed in Section 7.3.

The SAFDLs on linear heat rate and DNBR are intended to enforce the principal thermal-hydraulic design bases given in Subsection 4.4.1 (i.e., the avoidance of thermally induced fuel damage during normal steady-state operations and AOOs).
3.1.17 Criterion 21 – Protection System Reliability and Testability

“The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.”

Response:

The protection system is designed to conform with the requirements of IEEE Std. 603-1991 (Reference 6) and other standards, as noted in Subsection 7.1.2. No credible single failure will result in a loss of the protection function. The protection channels are independent with respect to wire routing, sensor mounting, and supply of power.

Each channel of the protection system, including the sensors up to the reactor trip switchgear system (RTSS) and ESFAS actuation devices, is capable of being checked during reactor operation. Process sensors of each channel in the protection systems are checked by comparison of the redundant process sensor values using the discrete indications and alarms in the MCR, as described in Subsection 7.7.1.2.

To minimize an inadvertent actuation of an ESF system or an inadvertent reactor trip, the protection systems use a coincidence of two-out-of-four logic to operate. The channel being testing is bypassed so the protection system converts to a two-out-of-three logic. This allows periodic testing and the operation of various protective functions without a loss of the protection function.

3.1.18 Criterion 22 – Protection System Independence

“The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional
diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.”

Response:

The protection systems conform with the independence requirements of IEEE Std. 603-1991. Four independent measurement channels, complete with sensors, sensor power supplies, signal conditioning units, and bistable trip functions, are provided for each protective parameter monitored by the protection systems, except for the CEA position sensors that are two-fold redundant. The measurement channels are provided with a high degree of independence by separate connection of the channel sensors to the process systems. Refer to Chapter 7 for a more detailed description of the protection systems.

Power to the protection system channels is provided by independent vital bus power supply systems. The power supply systems are described in Subsection 8.3.2.

Functional diversity is incorporated into the system design to prevent a loss of the protective function. When an RPS trip function is required, it is frequently backed up by other trip functions. The ESFAS actuation signals are used to actuate two or four independent ESF trains.

The diverse protection system augments reactor trip and auxiliary feedwater system (AFWS) actuation by using separate and diverse non-Class 1E trip logic from that used by the plant protection system.

The design goals are accomplished without excessive complexity by using only four channels for each parameter. This allows for testing and maintenance of a channel without reducing the system to a single channel for trip, which would make the system susceptible to spurious trip or actuation signals.

Environmental and seismic qualifications are also performed using type tests, specific equipment tests, appropriate analyses, or prior operating experience. Further information is provided in Sections 3.10 and 3.11.

3.1.19 Criterion 23 – Protection System Failure Modes

“The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse
environments (e.g., extreme heat or cold, fire, pressure, steam, water, radiation) are experienced.”

Response:

The plant protection system trip channels are designed to fail into a safe state or into a state established as acceptable in the event of loss of power supply. A failure is assumed to occur in only one channel (i.e., a single failure). This channel can be placed into bypass, which places the RPS/ESFAS local coincidence logic into a two-out-of-three configuration. Refer to Sections 7.2 and 7.3 for information on the failure modes and effects analysis.

A loss of power to CEDM coils would cause the CEAs to insert into the core. Redundancy, channel independence, and separation are incorporated into the protection system design to minimize the possibility of the loss of a protective function. The loss of offsite power would cause the standby diesel electric generators to start and energize the ESF trains with actuation signals present.

3.1.20 Criterion 24 – Separation of Protection and Control Systems

“The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.”

Response:

Protection system components and control system components are electrically and functionally isolated from each other. Further details are provided in Sections 7.2, 7.3, and 7.7.

Isolation devices provide reasonable assurance that when protection signals are used by non-safety systems and non-safety signals are used by safety systems, credible single failures in the non-safety system do not degrade the performance of the safety system. In addition to the electrical and physical isolation, functional isolation between non-safety systems and safety systems is provided. The functional isolation is provided by priority logic in the safety systems or by signal selector logic in the non-safety systems. The
priority logic provides reasonable assurance that safety actuation signals, both automatic and manual (system level and component level), override all control signals from the non-safety systems. Signal selection logic in the control system prevents erroneous control actions due to single sensor failures. Eliminating these erroneous control actions prevents challenges to the protection system while it is degraded due to the same sensor failure.

The adequacy of the system isolation capability has been verified by testing under conditions of postulated credible faults. The failure of any single control system component or channel, or the failure or removal from service of any single protection system component or channel, which is common to the control and protection system, leaves intact a system that satisfies the requirements of the protection system.

3.1.21 Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions

“The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.”

Response:

Shutdown of the reactor is accomplished by the opening of the reactor trip switchgear system (RTSS) circuit breakers, which interrupts power to the CEDM coils. Actuation of the circuit breakers is independent of any existing control signals.

The protection systems are designed such that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, including the withdrawal of a single full- or part-strength CEA. Analyses of possible reactivity control system malfunctions are described in Chapter 15.

3.1.22 Criterion 26 – Reactivity Control System Redundancy and Capability

“Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of
reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.”

Response:

Two independent reactivity control systems of different design principles are provided. The first system, using CEAs, includes a passive means (gravity) for inserting CEAs and is capable of reliably controlling reactivity changes to provide reasonable assurance that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. The CEAs can be mechanically driven into the core.

The appropriate margin for stuck rods is provided by assuming in the analyses of AOOs that the highest-worth CEA does not fall into the core.

The second system, using neutron-absorbing soluble boron, is capable of reliably compensating for the rate of reactivity changes resulting from planned normal power changes (including xenon burnup) such that SAFDLs are not exceeded. This system is capable of holding the reactor subcritical under cold conditions.

Either system is able to insert negative reactivity at a rate sufficient to prevent exceeding SAFDLs as the result of a power change (i.e., positive reactivity added by xenon burnup).

3.1.23 Criterion 27 – Combined Reactivity Control Systems Capability

“The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.”

Response:

Dissolved boron addition capability provided by the safety injection system (SIS) (Chapter 6) in conjunction with the CEAs is such that under postulated accident conditions (Chapter 15), even with the CEA of highest worth stuck out of the core, adequate reactivity control is available to maintain short- and long-term core cooling.
3.1.24 Criterion 28 – Reactivity Limits

“The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.”

Response:

The bases for CEA design include providing reasonable assurance that the reactivity worth of any one CEA is not greater than a preselected maximum value. The CEAs are divided into two sets, a shutdown set and a regulating set, further subdivided into groups as necessary. Administrative procedures and interlocks provide reasonable assurance that only one group is withdrawn at a time, and that regulating groups are withdrawn only after shutdown groups are fully withdrawn. The regulating groups are programmed to move in sequence and within limits that prevent the rate of reactivity addition and the worth of individual CEAs from exceeding limiting values.

The maximum rate of reactivity addition that can be produced by the chemical and volume control system (CVCS) is too low to induce pressure forces significant enough to rupture the RCPB or disturb the reactor vessel internals.

The RCPB (Chapter 5) and reactor internals (Chapter 4) are designed to appropriate codes (e.g., the response to Criterion 14) and accommodate the static and dynamic loads associated with an inadvertent, sudden release of energy such as that resulting from a CEA ejection or steam line break (Chapter 15) without rupture and with limited deformation so that core cooling capability is not impaired.

3.1.25 Criterion 29 – Protection Against Anticipated Operational Occurrences

“The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.”
Response:

Plant events, designated in ANSI/ANS 51.1, have been carefully considered in the design of the protection and reactivity control systems. Redundancy, independence, and testability have been considered in the design. These considerations, coupled with careful component selection, overall system testing, and adherence to detailed quality assurance requirements, provide reasonable assurance that protection and reactivity control systems will accomplish their safety functions in the event of AOOs.

Refer to Chapter 7 for detailed descriptions of the protection systems and Chapter 17 for description of design quality assurance. DBE analysis is addressed in Chapter 15.

3.1.26 Criterion 30 – Quality of Reactor Coolant Pressure Boundary

“Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.”

Response:

RCPB components are designed, fabricated, erected, and tested in accordance with ASME Section III. All components are classified safety Class 1 or 2 in accordance with ANSI/ANS 51.1 and undergo all quality assurance measures that are appropriate for the respective classification.

Means are provided for identifying the source of reactor coolant leakage, which includes detection of leakage to systems connected to the RCPB as well as leakage from the boundary into the containment.

Instrumentation is provided to indicate and record makeup flow rates to the primary water system. This instrumentation permits detection of both suddenly occurring and gradually increasing leaks. Leakage detection methods are described in Subsection 5.2.5.1.

3.1.27 Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

“The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident
conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.”

Response:

All the RCPB components are designed and constructed in accordance with ASME Section III and conform with test and inspection requirements of these codes. The test and inspection requirements provide reasonable assurance that flaw sizes are limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel, on which tests and inspections exceeding ASME Code requirements are performed. The tests and inspections performed on the reactor vessel are described in Subsection 5.3.1.3.

Carbon and low-alloy steel materials that form part of the RCPB are tested in accordance with fracture toughness requirements for materials in ASME Section III. Nonductile failure prevention is reasonably assured by adhering to the relevant sections of the ASME Code.

Excessive embrittlement of the reactor vessel material due to neutron irradiation is prevented by providing an annulus of coolant water between the reactor core and the vessel. In addition, to minimize the effects of irradiation on material toughness properties of core beltline materials, restrictions on upper limits for residual elements that directly influence the reference temperature for nil-ductility transition (RTNDT) shift are required by design specification. Specifically, upper limits are placed on copper, nickel, phosphorous, sulfur, and vanadium.

The maximum integrated fast neutron exposure of the reactor vessel wall opposite the midplane of the core is less than $9.5 \times 10^{19}$ n/cm$^2$. This value assumes a 60-year design life and a 93 percent plant capacity factor. The maximum expected increase in transition temperature for vessel weld materials is approximately $31.4 \, ^\circ$C ($56.6 \, ^\circ$F) provided that weld metal is conservatively assumed to be exposed to the peak value of neutron fluence, $9.5 \times 10^{19}$ n/cm$^2$. Actual locations of welds are outside the active core region. The actual change in material toughness properties due to irradiation is verified periodically.
during plant lifetime by a material surveillance program. Based on the initial $RT_{NDT}$ of archive pressure vessel beltline material, which is $-23.3 \, ^\circ C \, (-10 \, ^\circ F)$ or less, operating restrictions are applied as necessary to limit vessel stresses.

The thermal stresses induced by the injection of cold water into the vessel following a LOCA have been examined. Analyses have shown that there is no gross yielding across the vessel wall when using the minimum specified yield strength in ASME Section III.

3.1.28 Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

“Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.”

Response:

Provisions have been made in the design for inspection, testing, and surveillance of the RCS boundary as required by ASME Section XI. The system designer recommends a reactor vessel surveillance program to the owner. The reactor vessel surveillance program provided to the owner conforms with 10 CFR Part 50, Appendix H. Sample pieces taken from the same material used in fabrication of the reactor vessel are installed between the core and the vessel inside wall. These samples are removed and tested by the owner at intervals during vessel life to provide an indication of the extent of the neutron embrittlement of the vessel wall. Charpy tests are performed on the samples to develop a Charpy transition curve. By comparing this curve with the Charpy curve and drop weight test results for specimens taken at the beginning of the vessel life, the change of $RT_{NDT}$ is determined and operating procedures adjusted as required. Further details are provided in Chapter 5.

3.1.29 Criterion 33 – Reactor Coolant Makeup

“A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power
is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.”

Response:

Reactor coolant makeup during normal operation is provided by the CVCS. The two charging pumps can be powered from either onsite or offsite power sources. The system is described in Subsection 9.3.4. The CVCS has the capability of replacing the flow loss to the containment due to leaks in small reactor coolant lines such as instrument and sample lines. These lines have 5.56 mm (7/32 in.) diameter by 25.4 mm (1 in.) long flow-restricting devices to limit loss of RCS coolant due to postulated pipe breaks in RCS piping.

The CVCS is not required to perform any safety-related function, such as accident mitigation or safe shutdown. This does not, however, compromise the defense-in-depth provided by the system as the normal means of maintaining RCS inventory and primary water chemistry. The CVCS is essential to day-to-day plant operations and has been provided with a high degree of reliability and redundancy, and is designed in accordance with accepted industry standards and quality assurance commensurate with its importance to plant operation. Design criteria, including ASME Code classification assignments, are in accordance with ANSI/ANS 51.1, which requires that portions of the CVCS in the RCPB, all portions that provide reasonable assurance of containment isolation and reactor coolant normal makeup, have a rigorous safety classification in accordance with the functional performance requirements.

3.1.30 Criterion 34 – Residual Heat Removal

“A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”
Response:

Residual heat removal capability is provided by the shutdown cooling system (SCS) for reactor coolant temperature and pressure less than 176.7 °C (350 °F) and 31.6 kg/cm²A (450 psia). For temperatures greater than 176.7 °C (350 °F), this function is provided by the SGs. The AFWS provides a dedicated, independent, safety-related means of supplying feedwater (FW) to the SGs for removal of heat and prevention of reactor core uncovery. The design incorporates sufficient redundancy, interconnections, leak detection, and isolation capability to provide reasonable assurance that the residual heat removal function can be accomplished, assuming a single active failure. Within appropriate design limits, either system can remove fission product decay heat at a rate that SAFDLs and the design conditions of the RCPB are not exceeded.

The SCS and SG auxiliaries are designed to operate from offsite or from onsite power sources.

Further description of the SCS is provided in Subsection 5.4.7, and further description of the steam and power conversion system is provided in Chapter 10.

3.1.31 Criterion 35 – Emergency Core Cooling

“A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

Response:

Emergency core cooling is provided by the SIS as described in Section 6.3. The system is designed to provide cooling water to limit peak clad temperature to less than 1,204 °C (2,200 °F), to remove heat at a rate sufficient to maintain the fuel in a coolable geometry
and to provide reasonable assurance that zirconium-water reaction is limited to a negligible amount (less than 1 percent) following a LOCA, including breaks in the pipe in the reactor coolant pressure boundary up to and including a break size equivalent to the double-ended rupture of the largest pipe in the RCS. Performance of the emergency core cooling system (ECCS) has been analyzed using the evaluation models complying with the requirements of 10 CFR 50.46(a)(1), and performance has been verified adequate to meet the criteria prescribed in 10 CFR 50.46(b).

The system design includes provisions to provide reasonable assurance that the required safety functions are accomplished with either onsite or offsite electrical power, assuming a single failure of any component which is qualified as described below.

The single failure considered may be limited to an active failure\(^1\) during the short-term cooling phase of safety injection or an active or limited-leakage passive failure\(^2\) during the long-term cooling phase of safety injection.

Although the SIS is designed to accommodate a limited leakage passive failure during the long-term cooling phase, it does not accommodate arbitrary large-leakage passive failures, such as the complete double-ended severance of piping, which are extremely low probability events.

3.1.32 Criterion 36 – Inspection of Emergency Core Cooling System

“The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.”

\(^1\) An active failure is a malfunction, excluding passive failure, of a component that relies on mechanical movement to complete its intended function upon demand. Examples of active failures include the failure of a valve to move to its correct position, or the failure of a pump, fan, or diesel generator to start. Spurious action of a powered component originating within the actuation system or its supporting systems shall be regarded as an active failure, unless specific design features or operating restrictions preclude such spurious action.

\(^2\) A passive failure is a failure of a component to maintain its structural integrity or the blockage of a process flow path. Blockage of a process flow path could occur, for example, due to separation of a valve disc from its stem. Refer to Subsection 6.3.2.5.4 for a description of a limited-leakage passive failure of the SIS.
Response:

The SIS is designed to facilitate access to all critical components. All pumps, heat exchangers, valves, and piping external to the containment structure are readily accessible for periodic inspection to provide reasonable assurance of system leak-tight integrity. Valves, piping, and tanks inside the containment may be inspected for leak-tightness during plant shutdowns for refueling and maintenance.

Reactor vessel internal structures, reactor coolant piping, and water injection nozzles are designed to permit visual inspection for wear due to erosion, corrosion, or vibration and nondestructive inspection techniques where applicable and desirable.

Descriptions of the inspection program are provided in Chapters 5, 6, and 16.

3.1.33 Criterion 37 – Testing of Emergency Core Cooling System

“The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.”

Response:

The SIS is provided with testing capability to demonstrate system and component operability. Testing can be conducted during normal plant operation with the test facilities arranged not to interfere with the performance of the systems or with the initiation of control circuits, as described in Section 6.3 and Chapter 14. During in-service testing, testing is performed to confirm that the SI pump return line to the in-containment refueling water storage tank (IRWST) allows each SI pump to be operated at rated flow.

3.1.34 Criterion 38 – Containment Heat Removal

“A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other
associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

Response:

The containment spray system (CSS) consists of two independent divisions. Each division consists of a containment spray pump, a containment spray heat exchanger, a containment spray pump miniflow heat exchanger, a containment spray header, and associated piping and valves. The heat removal capacity of each division is sufficient to keep the containment pressure and temperature below design conditions for any break size in the RCS piping up to and including a double-ended break of the largest reactor coolant pipe, with an unobstructed discharge from both ends.

The CSS takes suction borated water from the IRWST, which is located in the containment. Borated water is sprayed downward by the CSS from the upper regions of the containment to cool the containment atmosphere. The removed heat is transferred by the component cooling water via the CSS heat exchangers to the ultimate heat sink (UHS).

Suitable redundancy in components and features is designed into the CSS to maintain the pressure and temperature conditions below containment design conditions even in the event of a single failure including the loss of onsite or offsite electrical power.

The CSS is supplied from separate Class 1E buses and receives electrical power from electrically independent and redundant emergency power supply systems as well as offsite power supplies.

The CSS is described further in Subsections 6.2.2 and 6.5.2. Electrical power supplies are described further in Subsections 8.2 and 8.3.
3.1.35  Criterion 39 – Inspection of Containment Heat Removal System

“The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.”

Response:

Inspections of the CSS are conducted to confirm the integrity and capability and provide reasonable assurance of the system. All essential equipment of the CSS is located outside the containment, except for spray headers, nozzles, containment sump, IRWST, and associated piping. These components include two containment spray pumps, two containment spray pump miniflow heat exchangers, two containment spray heat exchangers, and independent containment spray headers.

The detailed arrangement and layout of system piping, pumps, heat exchangers, and valves provide the separation, availability, and accessibility required for periodic inspection. Nozzle inspection capability is provided as well.

The CSS is described further in Subsections 6.2.2 and 6.5.2.

3.1.36  Criterion 40 – Testing of Containment Heat Removal System

“The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.”

Response:

The containment heat removal system is arranged so that components can be tested periodically for operability. The operational tests are performed to verify the proper operation of CSS and include the calibration of instrumentation, verification of adequate pump performance, verification of the operability of all associated valves, and verification that the spray headers and spray nozzles are free of obstructions. Testing can be
conducted during normal plant operation with the system configured to not interfere with the performance of the initiation of control circuits, as described in Subsection 6.2.2.

All components of the CSS are hydrostatically tested in the manufacturer’s shop. A hydrostatic test is performed in the field to verify the leaktightness of the system.

The performance testing of containment spray pumps is conducted at some time other than refueling. The pumps are aligned to take suction from and return flow to the IRWST. Flow and head are recorded by the installed instrumentation.

Actuator-operated valves can be tested from the control room and operation verified by observing control room indication.

The valves on the inlets and outlets of the containment spray pumps can be tested to provide reasonable assurance that the valves operate properly.

3.1.37 Criterion 41 – Containment Atmosphere Cleanup

“Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.”

Response:

The CSS and containment hydrogen control system are provided to control fission products, hydrogen, oxygen, and other substances that may be released into the containment. The CSS is an automatically actuated ESF system that provides heat removal and fission product removal following a LOCA. The fission products are removed to reduce activity at the site boundary by the CSS and the post-accident pH of the sprayed fluid is controlled by using trisodium phosphate (TSP). The CSS consists of two independent divisions that
are supplied power from separate Class 1E buses and has sufficient redundancy to perform its safety functions. The CSS is designed so that a single failure coincident with a loss of offsite power does not prevent performance of the safety function.

Passive autocatalytic recombiners (PARs), which have 200 percent of required capacity, are located in containment and preclude hydrogen concentration buildup to detonable levels.

The systems are described in detail in Subsections 6.2.5 and 6.5.2.

3.1.38 Criterion 42 – Inspection of Containment Atmosphere Cleanup Systems

“The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.”

Response:

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically as required. Inspection of the CSS function relative to iodine removal is treated as described in GDC 39.

All major active components of the containment atmosphere cleanup systems are located outside containment and are readily accessible for periodic inspection.

The inspection and surveillance program for the containment hydrogen control system is described further in Subsection 6.2.5.

3.1.39 Criterion 43 – Testing of Containment Atmosphere Cleanup Systems

“The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.”
Response:

Testing of the CSS is conducted to provide reasonable assurance of structural and leaktight integrity and operability and performance in accordance with GDC 40. In addition, performance testing is conducted on CSS components. These tests are normally conducted while the plant is operating. System design includes provisions to prevent accidental containment spray during component testing. The test for the CSS is described further in Subsections 6.2.2 and 6.5.2.

The containment hydrogen control system is designed to permit periodic testing for confirming the continued operability. Testing may be conducted during normal plant operation or shutdown. The test for the containment hydrogen control system is described in Subsection 6.2.5 for details.

3.1.40  Criterion 44 – Cooling Water

“A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

Response:

The cooling water systems, which function to remove the combined heat load from safety-related systems and components under normal operating and accident conditions, are the component cooling water system (CCWS) and the essential service water system (ESWS). The CCWS is a closed-loop system that removes heat from nuclear safety-related and potentially radioactive systems. The ESWS removes heat from the CCWS and transfers it to the environment through the UHS.
Suitable redundancy, systems interconnection, leak detection, and isolation capabilities are incorporated into the design of these systems to provide reasonable assurance of the required safety function, assuming a single failure with available onsite or offsite power.

The CCWS and the ESWS are described in Subsections 9.2.2 and 9.2.1, respectively.

3.1.41 Criterion 45 – Inspection of Cooling Water System

“The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.”

Response:

Components of these systems are located in accessible areas with the exception of any underground piping for the ESWS. These components have suitable manholes, handholes, inspection ports, or other appropriate design and layout features to allow periodic inspection. The integrity of any underground piping is demonstrated during normal power operation as well as by pressure and functional tests. The inspection and testing requirements for the CCWS and the ESWS are described in Subsections 9.2.2 and 9.2.1, respectively.

3.1.42 Criterion 46 – Testing of Cooling Water System

“The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.”

Response:

The design provides for periodic testing of the cooling water systems for operability and functional performance.
The manufacturer is required to conduct performance tests of the components. An initial system flow test demonstrates proper functioning of the system. Thereafter, periodic tests provide reasonable assurance that components are functioning properly.

Cooling water system valves can be connected to the preferred power source at any time during reactor operation to demonstrate operability. The system is operated normally, thereby demonstrating operability. Remotely operated valves are exercised and actuation circuits are tested. Automatic actuation circuitry, valves, and pump breakers can also be checked when integrated system tests are performed during a planned cooldown of the RCS. Provisions have been made to permit periodic leakage tests to verify the continued leak-tight integrity of the systems. The inspection and testing requirements for the CCWS and the ESWS are provided in Subsections 9.2.2 and 9.2.1, respectively.

3.1.43  Criterion 50 – Containment Design Basis

“The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.”

Response:

The containment structure, including access openings and penetrations, is designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a LOCA up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment structure and ESF systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy, and for nuclear decay heat. The SIS is designed such that no single active failure could result in significant metal-water reaction, as described in Subsection 6.2.1.
The containment heat removal system is described in Subsection 6.2.2. Further information is provided in Subsections 3.8.1 and 3.8.2.

3.1.44 Criterion 51 – Fracture Prevention of Containment Pressure Boundary

“The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.”

Response:

The containment’s ferrite materials are selected to provide reasonable assurance that their temperature under normal operating and testing conditions is at least 16.7 °C (30 °F) above the nil-ductility transition temperature (NDTT). Detailed stress analyses of the containment liner anchors are conducted under normal and postulated accident conditions.

Further details regarding ferrite materials used in the containment and associated design requirements are described in Subsections 3.8.1, 3.8.2, and 6.1.1.

3.1.45 Criterion 52 – Capability for Containment Leakage Rate Testing

“The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.”

Response:

Criterion 52 is complied with the plant design. The provision for testing in conformance with this criterion and the provisions for conformance with 10 CFR Part 50, Appendix J, are described in Subsection 6.2.6.1. Containment design relative to test pressure is described in Subsection 3.8.1.3.
3.1.46 Criterion 53 – Provisions for Containment Testing and Inspection

“The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.”

Response:

The containment design permits access to penetrations and other important areas for implementation of the surveillance program described in Chapter 16. Penetrations with resilient seals and bellows are visually inspected and pressure tested periodically for leaktightness according to the Technical Specifications. Leakage rate testing is described in Subsection 6.2.6.

3.1.47 Criterion 54 – Piping Systems Penetrating Containment

“Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.”

Response:

Piping systems that penetrate containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection tests to provide reasonable assurance that the leakage is within the acceptable limits of 10 CFR Part 50, Appendix J, and Chapter 16.

The ESFAS circuitry provides the means for testing isolation valve operability.

The fuel transfer tube is not classified as a fluid system penetration. The blind flange and the portion of the transfer tube inside the containment are an extension of the containment boundary. The blind flange isolates the transfer tube at all times, except when the reactor is shut down for refueling. This assembly is a penetration in the same sense as the equipment hatches and personnel locks.
Subsection 6.2.4 addresses penetration design. Additional information is given in the responses to GDC 55, 56, and 57 (Subsections 3.1.48 through 3.1.50).

3.1.48 Criterion 55 – Reactor Coolant Pressure Boundary Penetrating Containment

“Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

a. One locked closed isolation valve inside and one locked closed isolation valve outside containment.

b. One automatic isolation valve inside and one locked closed isolation valve outside containment.

c. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

d. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.”
Response:

All RCPB lines penetrating containment meet the isolation criteria of GDC 55 using the following basis for specific lines in addition to those noted above:

a. Safety injection lines, as shown in Figure 6.3.2-1, are used to mitigate the consequences of accidents and therefore do not receive an automatic closure signal and are not locked closed.

b. When in shutdown cooling mode, the shutdown cooling system is an extension of the RCPB and is isolated from the environment by two isolation valves in series.

c. The charging and seal injection lines shown in Figure 9.3.4-1 have automatic valves outside containment that do not receive a closure signal (CIAS) because it is desirable to maintain charging and seal injection flow as long as the charging pumps are in operation.

d. Special cases are described in Subsection 6.2.4.

3.1.49 Criterion 56 – Primary Containment Isolation

“Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

a. One locked closed isolation valve inside and one locked closed isolation valve outside containment.

b. One automatic isolation valve inside and one locked closed isolation valve outside containment.

c. One locked-closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

d. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.”

Response:

Fluid systems conform with the requirements of GDC 56 with the following exceptions:

a. Lines that connect directly to the containment atmosphere and are used for mitigating the effects of accidents are connected to a closed piping system outside containment, which is isolated from the environment in accordance with the requirements of GDC 55.

b. In addition, the capability for remote double isolation at the containment boundary is provided in accordance with GDC 56.

c. Special cases are described in Subsection 6.2.4.

3.1.50 Criterion 57 – Closed System Isolation Valves

“Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.”

Response:

The system conforms with the requirements of GDC 57 as described in Subsection 6.2.4.

3.1.51 Criterion 60 – Control of Releases of Radioactive Material to the Environment

“The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental
conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.”

Response:

The sources and expected quantities of radioactive materials produced during normal reactor operation, including AOOs, are presented in Chapter 11. The radioactive waste systems to suitably control the release of these materials in gaseous and liquid effluents and handle radioactive solid wastes are described in Sections 11.2 through 11.4.

3.1.52 Criterion 61 – Fuel Storage and Handling and Radioactivity Control

“The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.”

Response:

The spent fuel cooling and cleanup system, fuel handling system, building ventilation systems for radiation control areas, and radioactive waste management systems are designed to provide reasonable assurance of adequate safety under normal and postulated accident conditions.

Structures, systems, and components are designed and located so that appropriate periodic inspection and testing can be performed.

Information on shielding is provided in Section 12.3. Radiation monitoring is described in Section 11.5 and Subsection 12.3.4.

The individual components that contain radioactivity are located in confined areas and ventilated through appropriate filtering systems. Information on radioactive waste management systems is provided in Chapter 11.
The spent fuel pool cooling and cleanup system provides cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed with redundant and testable features to provide reasonable assurance of continuous heat removal. The spent fuel pool cooling and cleanup system is described in Subsection 9.1.3.

The spent fuel pool is designed in accordance with seismic Category I requirements so that no postulated accident can cause excessive loss of coolant inventory.

3.1.53 **Criterion 62 – Prevention of Criticality in Fuel Storage and Handling**

“Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.”

**Response:**

The restraints and interlocks provided for safe handling and storage of new and spent fuel are described in Section 9.1.

The fuel handling area is laid out to preclude the spent fuel shipping cask from traversing the spent fuel storage pool.

The design of fuel storage racks provides reasonable assurance of an effective multiplication factor ($K_{\text{eff}}$). For new fuel storage racks, $K_{\text{eff}}$ is less than 0.95 with full density unborated water and less than 0.98 with optimum moderating condition. For spent fuel storage racks, $K_{\text{eff}}$ is less than 0.95 with partial credit for soluble boron and less than 1.0 with full density unborated water.

Criticality in the fuel storage area is prevented by the physical separation of fuel assemblies, the fixed neutron absorber attached on the spent fuel rack wall, and the presence of borated water in the fuel storage pool.

3.1.54 **Criterion 63 – Monitoring Fuel and Waste Storage**

“Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.”
Response:

Instrumentation is provided to monitor the spent fuel pool (SFP) water level and water temperature continuously, and their indications and annunciations are provided in the main control room (MCR). Two redundant safety-related fuel handling area radiation monitors are provided to monitor SFP area radioactivity continuously, and their indications and annunciations are provided in the MCR. In case of an inadvertent release of radioactivity, the area radiation monitor generates local and MCR alarms and an ESFAS fuel handling area emergency ventilation actuation signal as required for the appropriate safety actions. Information on the ESFAS is provided in Section 7.3 and information on the fuel storage area radiation monitoring system is provided in Subsection 9.4.2.

3.1.55 Criterion 64 – Monitoring Radioactivity Releases

“Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.”

Response

The reactor containment atmosphere is continuously sampled and monitored for any radioactivity that is released from normal operations (including AOOs) by a continuous air monitoring system located outside the containment. A separate system is provided to enable collection and analysis of grab samples of the containment atmosphere during normal operations and accident conditions. Area monitors such as high-range ion chambers are located inside the containment to measure radiation levels from normal and accident conditions.

Gamma and beta scintillators are located outside the containment for containment air monitoring.

Spaces containing components for recirculation of LOCA fluids and areas contiguous to the containment structure are monitored for airborne radioactivity by systems that sample and monitor the air exhausted from the associated areas. The systems consist of continuous air monitors, duct radiation monitors, and air samplers to enable the collection of samples of exhaust air for laboratory analysis during normal and accident conditions.
Effluent discharge paths and plant environs from the facility are continuously monitored for radioactivity during normal operations with continuous air and liquid monitoring systems. Sampling provisions are included to allow sample collection for analysis during normal operations and accident conditions. Extended-range noble gas monitors are provided to allow for continuous monitoring during accident conditions. The systems provided for monitoring radioactive releases from the facility are described in Sections 11.5 and 12.3.

3.1.56 Combined License Information

No COL information is required with regard to Section 3.1.

3.1.57 References


3.2 Classification of Structures, Systems, and Components

The APR1400 structures, systems, and components (SSCs) are classified according to nuclear safety classification, quality groups, seismic category, 10 CFR Part 50, Appendix B, quality assurance program, and codes and standards.

Safety-related SSCs are defined in 10 CFR 50.2 as SSCs that are relied on to remain functional during and following design basis events to ensure the following:

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe shutdown condition; or

c. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable guideline exposures in 10 CFR 50.34(a)(1) or 10 CFR 100.11

The APR1400 SSCs that perform the above safety functions are classified as safety-related. The safety-related SSCs meet the quality assurance requirements of Appendix B of 10 CFR Part 50. SSCs that do not perform the above safety functions are classified as non-safety-related, and the requirements of 10 CFR Part 50, Appendix B, are not applied to these SSCs.

However, APR1400 SSCs that are important to safety but are not safety-related are additionally classified so that they are designed to the appropriate quality standards. The augmented quality assurance requirements for these SSCs, which are described in Chapter 17, are commensurate with the importance of their safety functions. The areas where these augmented quality controls are applied to SSCs important to safety are anticipated transient without scram (ATWS), station blackout, fire protection, seismic Category II SSCs, and risk-significant non-safety-related SSCs determined by the design reliability assurance program, which is described in Section 17.4.

This section provides the methodology that is used to classify APR1400 SSCs. Seismic classifications are described in Subsection 3.2.1, and quality group classifications are described in Subsection 3.2.2. The safety classifications are described in Subsection 3.2.3.
3.2.1 Seismic Classification

General Design Criterion (GDC) 2 (Reference 1) requires, in part, that nuclear power plant SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. Plant features, including foundations and supports, that are designed to remain functional in the event of a safe shutdown earthquake (SSE) or surface deformation are designed as seismic Category I. These plant features are the features that are necessary to provide reasonable assurance of (1) the integrity of the reactor coolant pressure boundary (RCPB), (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) and 10 CFR 52.79 or 10 CFR 100.11 (Reference 2).

The earthquake for which these plant features are designed is defined as the SSE in 10 CFR Part 100, Appendix A (Reference 3), and 10 CFR Part 50, Appendix S (Reference 4). The SSE is based on an evaluation of the maximum earthquake potential and produces the maximum vibratory ground motion for which SSCs important to safety are designed to remain functional.

The seismic classification used in APR1400 complies with the seismic criteria stated in NRC RG 1.29 to meet the requirements of both GDC 2 and 10 CFR Part 50, Appendix S. Radioactive waste management SSCs comply with the seismic design criteria specified in NRC RG 1.143 (Reference 6). Radioactive waste management SSCs are designed to meet the design basis loads including natural phenomena and internal/external man-induced hazard design criteria in accordance with NRC RG 1.143. Designing and constructing radioactive waste management SSCs to meet the requirements of GDC 61 and the guidance on seismic design and classification in NRC RG 1.143 provide reasonable assurance that SSCs that contain radioactivity are properly classified and that radiation exposure as a result of a seismic event will be as low as is reasonably achievable (ALARA). The radwaste safety classification is described in Table 3.2-1.

Instrument sensing lines and their supports are designed in accordance with the seismic design criteria of NRC RG 1.151 (Reference 7). Compliance with Positions C.2 and C.3 of NRC RG 1.151 provides reasonable assurance that the instrument sensing lines used to
actuate or monitor safety-related systems are appropriately classified and will be capable of withstanding the effects of the SSE.

Fire protection systems are designed in accordance with the seismic design criteria of NRC RG 1.189 (Reference 8). Compliance with Positions 3.2.1, 6.1.1.2, and 7.1 of NRC RG 1.189 provides reasonable assurance that the SSCs important to safety that are required to function during an SSE are properly classified as seismic Category I, will function during such events, and will provide reasonable assurance that the safety functions can be performed.

Seismic categories are designated as seismic Category I, II, or III.

The seismic categories of SSCs are listed in Table 3.2-1. The COL applicant is to identify the seismic classification of site-specific SSCs that are to be designed to withstand the effects of the SSE (COL 3.2(1)).

The seismic category portions of SSCs are indicated by the class breaks shown on the flow diagrams for the appropriate systems described in this DCD. Seismic Category I requirements extend to the first seismic anchor beyond the interface of the classification change. Supports for piping and components have the same seismic classifications as the piping and components that are supported.

Seismic Category I, II, and III SSCs are defined as follows:

a. Seismic Category I

SSCs that are important to safety and designed to remain functional in the event of an SSE are designated as seismic Category I.

The selection of seismic Category I SSCs is in accordance with the definition above and the guidance provided by NRC RG 1.29 (Reference 5). Seismic Category I components have a designated safety class in accordance with ANSI/ANS 51.1 (Reference 9; see Subsection 3.2.3). All components in Safety Classes 1, 2, and 3 are seismic Category I. The portions of SSCs that form an interface between seismic Category I and seismic Category II/III features are designed to seismic Category I requirements in accordance with Regulatory Position C.3 of NRC RG 1.29 (Reference 5). Seismic Category I design
requirements extend to the first seismic anchor (restraint) beyond the defined boundaries.

Portions of some non-safety-related systems (e.g., fire protection system) are classified as seismic Category I to conform with NRC RG 1.189.

Seismic Category I SSCs are designed to remain functional and within the applicable stress and deformation limits (elastic range of material properties) when subjected to the effects of the vibratory motion of the operating basis earthquake (OBE) in combination with normal operation loads. This design is based on the design for SSE loads where an OBE is defined as one third of the SSE, as described in Subsection 3.7.1. Seismic Category I structures are protected from interaction with adjacent non-seismic structures, as described in Subsection 3.7.2.8. The seismic classifications of platforms and miscellaneous steel located in seismic Category I application are described in Subsection 3.8.3.

Seismic Category I SSCs meet the pertinent QA requirements of 10 CFR Part 50, Appendix B (Reference 10). The criteria used for the design of seismic Category I SSCs are described in Section 3.7.

b. Seismic Category II

Seismic Category II applies to SSCs which do not perform safety-related function, and whose continued function is not required, but whose structural failure could reduce the functioning of a seismic Category I SSC to an unacceptable safety level or could result in incapacitating injury to occupants of the control room.

Seismic Category II SSCs are designed to preclude a gross structural failure resulting from an SSE that could degrade the ability of an adjacent safety-related SSC to function to an unacceptable level or result in incapacitating injuries to personnel in the main control room (MCR).

Seismic Category II SSCs meet the augmented quality assurance requirements for non-safety-related SSCs as described in Section 17.5.

c. Seismic Category III
All SSCs not covered by seismic Category I or II are classified as seismic Category III and are designed in accordance with industry codes and standards as applicable for their design function.

3.2.2 System Quality Group Classification

GDC 1 of 10 CFR Part 50 Appendix A (Reference 1) requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Fluid systems and pressure-retaining components are part of the RCPB and other fluid systems that are important to safety. Such systems (1) prevent or mitigate the consequences of accidents and malfunctions originating in the RCPB, (2) permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) retain radioactive material.

NRC RG 1.26 (Reference 11) is the principal document that is used to identify, on a functional basis, the components of systems that are important to safety and that are in Quality Groups A, B, C, and D, which are defined below. ASME Section III, or Safety Class 1, components that are part of the RCPB are identified in 10 CFR 50.55a (Reference 12).

Systems and components are assigned to quality groups in accordance with the quality group classification system (NRC Quality Groups A, B, C or D) defined in NRC RG 1.26 (Reference 11), which was established for water-steam-containing components important to safety. Two other quality groups, E and G, are defined, in addition to those designated in NRC RG 1.26, to indicate the governing design codes for the components that are not covered under NRC Quality Groups A, B, C, or D.

The quality group classifications and codes and standards for mechanical and fluid systems and components are listed in Table 3.2-1 and are shown on the applicable flow diagrams. The flow diagrams identify the classification boundaries of interconnecting piping and valves, as well as the interfaces between the safety-related and non-safety-related portions of each system. The COL applicant is to identify the quality group classification of site-specific systems and components and their applicable codes and standards (COL 3.2(2)).

Quality Groups A through E and G are defined below.
Quality Group A

Quality Group A applies to the components that are part of the reactor coolant pressure boundary (RCPB), except for the portions that are included in Quality Group B below. This exclusion applies to components whose failure would not prevent the reactor from being shut down and cooled down in an orderly fashion with normal makeup and components that are or can be isolated from the reactor coolant system by two valves in series (with automatic closure of open valves).

Quality Group A water-, steam-, and radioactive-waste-containing components are designed to meet the requirements for Class 1 components in ASME Section III, Division I, NB.

Quality Group B

Quality Group B applies to water- and steam-containing components that support the systems or portions of systems listed in the regulatory position C.1 of NRC RG 1.26.

These systems or portions of systems are as follows:

a. Portions of the RCPB that are excluded from Quality Group A

b. Systems or portions of systems important to safety that are designed for the (i) emergency core cooling, (ii) post-accident containment heat removal, or (iii) post-accident fission product removal

c. Systems or portions of systems important to safety that are designed for (i) reactor shutdown or (ii) residual heat removal

d. Portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation

e. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure
Quality Group B water- and steam-containing components are designed to meet the requirements for Class 2 components in ASME Section III, Division I, NC and NG.

**Quality Group C**

Quality Group C applies to water-, steam-, and radioactive-waste-containing components that are not part of the reactor coolant pressure boundary or included in Quality Group B but part of the following:

a. Cooling water and auxiliary feedwater systems or portions of those systems important to safety that are designed for emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, or residual heat removal from the reactor and from the spent fuel storage pool

b. Cooling water and seal water systems or portions of those systems important to safety that are designed for the functioning of components and systems important to safety

c. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure

d. Systems, other than radioactive waste management systems, not covered by the above item a. through item c. that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body; only single component failures need be assumed for those systems located in Seismic Category I structures, and no credit should be taken for automatic isolation from other components in the system or for treatment of released material, unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component.

Quality Group C pressure retaining components are designed to meet the requirements for Class 3 components in ASME Section III, Division I, ND.
Quality Group D

Quality Group D applies to non-safety-related systems and components that are not covered under Quality Group A, B, or C and that are designed to ASME B31.1 (Reference 13) code criteria or other codes and standards listed in Table 1 of NRC RG 1.26. Quality Group D may include parts or portions of systems that contain or may contain radioactive material.

The radwaste management system and steam generator blowdown system (SGBS), which contain or may contain radioactive materials, are designed in accordance with applicable codes and standards, QA requirements, and guidance provided in NRC RG 1.143.

Quality Group E

Quality Group E pertains to non-safety-related fluid systems and components that are designed to codes other than ASME B31.1 code criteria and codes and standards listed in NRC RG 1.26.

Quality Group G

Quality Group G pertains to safety-related fluid systems and components that are designed to codes and standards other than those listed for Quality Group A, B and C in Table 1 of NRC RG 1.26.

3.2.3 Safety Class

Fluid system components important to safety are classified in accordance with ANSI/ANS 51.1 (Reference 9). Safety Class 1, 2, 3, and non-nuclear safety (NNS) of ANSI/ANS 51.1 are equivalent, on a functional basis, to Quality Groups A, B, C, D of NRC RG 1.26. The criteria establish safety classes that are used as a guide to the selection of codes, standards, and quality assurance provisions for the design and construction of the components. The safety class designations are also used as a guide to the fluid system components that are classified as seismic Category I and II (see Subsection 3.2.1).

The safety classification in ANSI/ANS 51.1 is summarized as follows:

a. Safety Class 1

Safety Class 1 (SC-1) applies to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a
loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Section III.

b. Safety Class 2

Safety Class 2 (SC-2) applies to pressure-retaining portions and supports of primary containment and other mechanical equipment, whose requirements are within the scope of the ASME Section III, which are not assigned to SC-1 but are relied on to accomplish the following safety functions:

1) Provide fission product barriers or primary containment radioactive material holdup or isolation

2) Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere (e.g., containment spray)

3) Introduce emergency negative reactivity to make the reactor subcritical (e.g., boron injection system), or restrict the addition of positive reactivity via pressure boundary equipment

4) Provide reasonable assurance of emergency core cooling where the equipment provides coolant directly to the core (e.g., residual heat removal, emergency core cooling)

5) Provide or maintain sufficient reactor coolant inventory for emergency core cooling

c. Safety Class 3

Safety Class 3 (SC-3) applies to equipment not included in SC-1 or SC-2 that is designed and relied on to accomplish the following safety functions:

1) Provide the functions defined in SC-2 where equipment, or portions thereof, is not within the scope of ASME Section III

2) Except for the primary containment boundary extension function, provide reasonable assurance of hydrogen concentration control of the primary containment atmosphere to acceptable limits
3) Remove radioactive material from the atmosphere of confined space outside primary containment (e.g., MCR, fuel building) containing SC-1, SC-2, or SC-3 equipment

4) Introduce negative reactivity to achieve or maintain subcritical reactor conditions (e.g., boron makeup)

5) Provide or maintain sufficient reactor coolant inventory for core cooling (e.g., reactor coolant normal makeup system)

6) Maintain geometry within the reactor to provide reasonable assurance of core reactivity control or core cooling capability (e.g., core support structures)

7) Structurally load-bear or protect SC-1, SC-2, or SC-3 equipment

8) Provide radiation shielding for the MCR or offsite personnel

9) Provide reasonable assurance of required cooling for liquid-cooled stored fuel (e.g., SFP and cooling system)

10) Provide reasonable assurance of nuclear safety functions provided by SC-1, SC-2, or SC-3 equipment (e.g., provide heat removal for SC-1, SC-2, or SC-3 heat exchangers, provide lubrication of SC-1, SC-2, or SC-3 pumps, provide fuel oil to emergency diesel engine)

11) Provide actuation or motive power for SC-1, SC-2, or SC-3 equipment

12) Provide information or controls to provide reasonable assurance of capability for manual or automatic actuation of nuclear safety functions required of SC-1, SC-2, or SC-3 equipment

13) Supply or process signals or supply power required for SC-1, SC-2, or SC-3 equipment to perform its required nuclear safety functions

14) Provide a manual or automatic interlock function to provide reasonable assurance that the proper performance of nuclear safety functions required of SC-1, SC-2, and SC-3 equipment is maintained
15) Provide an acceptable environment for SC-1, SC-2, or SC-3 equipment and operating personnel

d. Non-nuclear safety

Non-nuclear safety (NNS) applies to equipment or structures that are not included in SC-1, SC-2, or SC-3. These items are not relied on to perform a nuclear safety function.

All pressure-containing components in SC-1, SC-2, and SC-3 are designed, manufactured, and tested in accordance with the requirements of ASME Section III Code Class 1, 2, and 3, respectively. Components designated as NNS are designed and constructed with appropriate consideration of the intended service using applicable industry codes and standards. Table 3.2-2 shows the interrelationship and/or correspondence between the classifications.

The electrical equipment in SC-3 is in accordance with IEEE Std. 308 (Reference 14), IEEE Std. 603 (Reference 15), and applicable IEEE standards. The structures in SC-2 and SC-3 are in accordance with ASME Section III, Division 1, MC (Reference 16); ASME Section III, Division 2, CC (Reference 17); ACI-349 (Reference 18); and ANSI/AISC N690 (Reference 19).

The safety classification system is also used to identify the components that fall under the requirements of 10 CFR Part 50, Appendix B (Reference 10). Components in safety Classes 1, 2, and 3 are designed and manufactured under a rigorous quality assurance program reflecting the requirements of 10 CFR Part 50, Appendix B, and are designated as such in Table 3.2-1 in the column labeled 10 CFR Part 50, Appendix B. Components that do not serve a safety-related function are not subject to the quality assurance requirements of 10 CFR Part 50, Appendix B, and are designated as such in Table 3.2-1 in the column labeled 10 CFR Part 50, Appendix B.

Supports for piping and components have the same classification as the component or piping supported. Supports for APR1400 Quality Group A, B, and C mechanical components and piping are constructed to ASME Section III, Subsection NF requirements. The principal construction code for supports for non-safety-related components and piping is the same as that for the supported component or piping.
The safety classification systems described above meet the intent of NRC RG 1.26 and the requirements of 10 CFR 50.55a.

3.2.4 Classification Listings

Table 3.2-1 provides component classifications as defined in Subsections 3.2.1 through 3.2.3. Table 3.2-1 also provides the quality assurance requirements of 10 CFR Part 50, Appendix B, and the applicable codes and standards.

3.2.5 Combined License Information

COL 3.2(1) The COL applicant is to identify the seismic classification of site-specific SSCs that should be designed to withstand the effects of the SSE.

COL 3.2(2) The COL applicant is to identify the quality group classification of site-specific systems and components and their applicable codes and standards.

COL 3.2(3) The COL applicant is to provide the classification of structures, systems, and components for turbine generator building drain system.

3.2.6 References


2. 10 CFR 100.11, “Determination of Exclusion Area, Low Population Zone, and Population Center Distance,” U.S. Nuclear Regulatory Commission.


17. ASME Section III, Division 2, Subsection CC “Concrete Containments,” American Society of Mechanical Engineers, the 2001 Edition with the 2003 Addenda.


### APR1400 DCD TIER 2

**Classification of Structures, Systems, and Components (1)**

<table>
<thead>
<tr>
<th>SSC Identification</th>
<th>Location (2)</th>
<th>Safety Class</th>
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<td><strong>I. Major Structures</strong></td>
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<td>4. Turbine Generator Building</td>
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<td>7. Alternate Alternating Current Gas Turbine Generator Building</td>
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#### II. Systems and Components

1. AC – Auxiliary Process Cabinet
   a. APC A/B/C/D
      
      | Location(2) | Safety Class | Quality Group | Codes and Standards                                                                 | 10 CFR 50, App. B(3) | Seismic Category | Remarks |
      |-------------|--------------|---------------|-------------------------------------------------------------------------------------|----------------------|------------------|---------|
   b. APC N1/N2
      
      | Location(2) | Safety Class | Quality Group | Codes and Standards                                                                 | 10 CFR 50, App. B(3) | Seismic Category | Remarks |
      |-------------|--------------|---------------|-------------------------------------------------------------------------------------|----------------------|------------------|---------|
      | AB          | NNS          | N/A           | N/A                                                                                 | A                    | III              |

2. AF – Auxiliary Feedwater
   a. Auxiliary feedwater pumps
      1) Pumps
      
      | Location(2) | Safety Class | Quality Group | Codes and Standards                                                                 | 10 CFR 50, App. B(3) | Seismic Category | Remarks |
      |-------------|--------------|---------------|-------------------------------------------------------------------------------------|----------------------|------------------|---------|
      | AB          | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda                                          | Yes                  | I                |
   b. Auxiliary feedwater pump suction piping and valves from auxiliary feedwater suction manual valves (AFW-V1001 A/B, AFW-V1002 A/B)
      
      | Location(2) | Safety Class | Quality Group | Codes and Standards                                                                 | 10 CFR 50, App. B(3) | Seismic Category | Remarks |
      |-------------|--------------|---------------|-------------------------------------------------------------------------------------|----------------------|------------------|---------|
      | AB          | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda                                          | Yes                  | I                |
   c. Auxiliary feedwater pump discharge piping and valves up to and including emergency cooling water injection line manual valve (AFW-V2102 A/B) but excluding auxiliary feedwater isolation valves (AFW-V043 ~ 046)
      
<pre><code>  | Location(2) | Safety Class | Quality Group | Codes and Standards                                                                 | 10 CFR 50, App. B(3) | Seismic Category | Remarks |
  |-------------|--------------|---------------|-------------------------------------------------------------------------------------|----------------------|------------------|---------|
  | AB          | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda                                          | Yes                  | I                |
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<tr>
<td>d. Auxiliary feedwater pump discharge piping from auxiliary feedwater isolation valves (AFW-V043 ~ 046) up to feedwater connection</td>
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<td>SC-2</td>
<td>B</td>
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<td>e. Auxiliary feedwater pump recirculation piping and valves up to and including auxiliary feedwater recirculation isolation valves (AFW-V1011 A/B, AFW V1013 A/B)</td>
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<td>g. Non-safety-related piping/equipment in safety-related areas</td>
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<td>h. Other non-safety-related piping/equipment in non-safety-related areas</td>
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3. AN – Alarm

a. Qualified indication and alarm-non-safety (QIAS-N) alarm | AB | NNS | N/A | N/A | A | I | (N-9) |

b. Information processing system (IPS) alarm | AB | NNS | N/A | N/A | A | II | (3)(d) |

4. AS – Auxiliary Steam


b. Non-safety-related piping and components in safety-related areas | AB, RCB | NNS | D | ASME B31.1-2010 | A | II | (3)(d) |

c. Others | TGB, CB | NNS | D | ASME B31.1-2010 | N/A | III |
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<td>a. Auxiliary feedwater pump turbines</td>
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<td>b. Steam admission/exhaust/ preheating lines and valves</td>
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<td>c. Non-safety-related piping/component in safety-related areas</td>
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<td>d. Others</td>
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<td>b. Auxiliary feedwater makeup piping up to and including auxiliary feedwater storage tank inlet manual valves (AX-V1605, AX-V1606)</td>
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<td>c. Auxiliary feedwater makeup piping from auxiliary feedwater storage tank inlet manual valves (AX-V1605, AX-V1606) up to and including auxiliary feedwater storage tank makeup check valve (AX-V1600)</td>
<td>AB</td>
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<td>d. Auxiliary feedwater pump suction piping and valves including emergency cooling water injection line manual valves (AX-2679A/B) from auxiliary feedwater storage tanks up to and excluding auxiliary feedwater suction manual valves (AFW-V1001 A/B, AFW-V1002 A/B)</td>
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<td>e. Component cooling water / essential chilled water makeup piping up to and including component cooling water / essential chilled water makeup pump supply valves (AX-V1607, AX-V1608)</td>
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<td>f. Auxiliary feedwater storage tank sampling piping up to and including auxiliary feedwater storage tank grab sample test valves (AX-V2642, AX-V2644)</td>
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<td>g. Auxiliary feedwater storage tank overflow piping</td>
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<td>h. Auxiliary feedwater storage tank drain piping from auxiliary feedwater storage tank drain valves (AX-V2641, AX-V2643) to condensate polishing area sump in safety-related area</td>
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<td>i. Auxiliary feedwater storage tank drain piping up to condensate polishing area sump in non-safety-related area</td>
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<td>j. Auxiliary feedwater storage tank cross connection line up to and including AFWST connection manual valves (AX-V1621, AX-V1622)</td>
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<td>k. Auxiliary feedwater storage tank cross connection line between and excluding AFWST connection manual valves (AX-V1621, AX-V1622)</td>
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<td>l. Non-safety backup supply line up to and including AFW pump suction manual valves (AX-V1623, AX-V1624)</td>
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<td>m. Non-safety backup supply line from AFW pump suction manual valves (AX-V1623, AX-V1624) up to and including raw water supply valve (AX-V1208) and condensate storage tank water supply valve (AX-V1627)</td>
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7. BI – Bypass and Inoperable Status Indication

| a. Control logic and indication device | NNS | N/A | IEEE Std. 603-1991 | A | II (3)(d), (8) |

8. CA – Condenser Vacuum

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<td>b. Non-safety-related piping and components in safety-related areas</td>
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<td>c. Condenser vacuum pumps, booster fan and other components</td>
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<td>NNS</td>
<td>D</td>
<td>ASME B31.1-2010, HEI Standards-2006, ASME AG-1-2009</td>
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**9. CC – CCW**

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<td>c. CCW makeup pumps</td>
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<td>d. CCW surge tanks</td>
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<td>e. Chemical addition tank</td>
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<td>f. Component cooling water supply and return piping and valves excluding the following 1) through 9) below:</td>
<td>AB, CCWHXB, EDGB, Yard</td>
<td>SC-3</td>
<td>C</td>
<td>ASME Section III ND-2007 with 2008 addenda</td>
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<td>1) Containment penetration piping of RCP cooler supply line between and including the valves, CC-231 and CC-1099 in the division I</td>
<td>RCB</td>
<td>SC-2</td>
<td>B</td>
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<td>2) Containment penetration piping of RCP cooler return line between and including the valves, CC-249, CC-250, and CC-1100 in the division I</td>
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<td>3) RCP cooler supply and return piping between the valves, CC-1099, CC-249, and CC-1100 in the division I</td>
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<td>4) Non-essential supply and return piping between the valve CC-145 and CC-147 in the division I excluding the following 5) through 7) below:</td>
<td>AB</td>
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<td>5) Containment penetration piping of letdown heat exchanger supply line between and including the valves CC-296, CC-297, and CC-1685 in the division I</td>
<td>RCB</td>
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<td>6) Containment penetration piping of letdown heat exchanger return line between and including the valve CC-301, CC-302, and CC-1686 in the division I</td>
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<td>7) Letdown heat exchanger supply and return piping between the valves, CC-297, CC-301, CC-1685, and CC-1686 in the division I</td>
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<td>8) Non-essential supply and return piping between the valve CC-146 and CC-148 in the auxiliary building of the division II</td>
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<td>9) Non-essential supply and return piping in the compound building of the division II</td>
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#### 10. CD – Condensate

| a. Piping in auxiliary bldg. | AB       | NNS          | D             | ASME B31.1-2010                      | A                     | II               | (3)(d)  |
| b. Condenser, condensate pumps, tanks, valves, strainers | TGB      | NNS          | D             | ASME B31.1-2010                      | N/A                   | III              |         |
| c. Deaerator storage tank   | TGB      | NNS          | D             | ASME Section VIII - 2007 with 2008 addenda | N/A                   | III              |         |
| d. Feedwater Heaters        | TGB      | NNS          | D             | ASME Section VIII - 2007 with 2008 addenda | N/A                   | III              |         |
| e. Other piping             | TGB      | NNS          | D             | ASME B31.1-2010                      | N/A                   | III              |         |

#### 11. CE – Control Element Assembly Drive

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<td>3) Extension shaft assembly</td>
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<td>b. Reactor trip switchgear</td>
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<td>N/A</td>
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<td>c. CEDM motor generator set</td>
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<td>ASME B16.5-2009</td>
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<td>aj. Letdown orifices</td>
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<td>1) Within RCPB</td>
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<td>2) Letdown, charging, seal injection, and aux. spray piping and valves</td>
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<td>(b) Letdown piping and valves from CV-523 outlet to CV-520 outlet</td>
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<td>(c) Letdown piping and valves from CV-520 outlet to CV-415 inlet</td>
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<td>N/A    III CV-870/894/895/896 and associated branch lines</td>
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<td>(d) Letdown piping and valves from CV-415 inlet to VCT inlet</td>
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<td>(e) RCP CBO piping and valves from CV-505 outlet to VCT inlet</td>
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<td>(f) RCP seal injection piping and valves form seal injection tee to CV-255 inlet</td>
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<td>(g) Charging piping and valves from VCT outlet to CV-524 inlet</td>
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<td>4) Reactor water drain collection</td>
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<td>(a) Inside containment (reactor drain tank)</td>
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<td>(b) Outside containment (equipment drain tank, reactor drain pump suction and discharge to holdup tank)</td>
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<td>D</td>
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<td>II</td>
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<tr>
<td>5) Boric acid recovery system (holdup tank to boric acid storage tank and reactor makeup water tank)</td>
<td>Yard</td>
<td>NNS</td>
<td>D</td>
<td>ASME B31.1-2010</td>
<td>A</td>
<td>II</td>
<td>(3)(d), CV-686/127: Seismic Cat. II</td>
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**Notes:**
- (2) Location codes: AB, RCB, Yard
- (3) Codes and Standards: 10 CFR 50, App. B
- (3)(d) Seismic Cat. II: CV-686/127

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<td>6) Boric acid supply (BAST to VCT/charging pump suction)</td>
<td>Yard, AB</td>
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<td>C</td>
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**21. CW – Circulating Water**

- **a.** [[CW pumps]]
  - CWPH
  - NNS
  - D
  - HI Standards – 2010
  - N/A
  - III

- **b.** Butterfly valves
  - TGB, CWPH
  - NNS
  - D
  - AWWA C504-2010
  - N/A
  - III

- **c.** Condenser tube cleaning system components
  - TGB
  - NNS
  - D
  - ASME B31.1-2010
  - N/A
  - III

- **d.** Circulating water pump lube water booster pumps
  - CWPH
  - NNS
  - D
  - HI Standards-2010
  - N/A
  - III

- **e.** [[Makeup pumps]]
  - Yard
  - NNS
  - D
  - HI Standards-2010
  - N/A
  - III

- **f.** [[Blowdown pumps]]
  - Yard
  - NNS
  - D
  - HI Standards-2010
  - N/A
  - III

- **g.** [[Cooling towers (including cooling tower fans)]]
  - Yard
  - NNS
  - D
  - ASME PTC 23-2003
  - N/A
  - III

- **h.** Piping and valves
  - TGB, CWPH, Yard
  - NNS
  - D
  - ASME B31.1-2010
  - N/A
  - III
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<td>f. Radwaste control room engineering workstation</td>
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22a. DA - Alternate AC Gas Turbine Generator System

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<tr>
<td>a. Gas turbine generator package including auxiliary subsystem</td>
<td>AAC GTGB</td>
<td>NNS</td>
<td>E</td>
<td>Manufacturer’s standard</td>
<td>A</td>
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<td>b. AAC diesel fuel oil tanks and day tanks</td>
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<td>NNS</td>
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<td>ASME Section VIII-2007 with 2008 addenda/API-650-2007 with 2008 addenda</td>
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<td>c. AAC diesel fuel oil transfer pump</td>
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<td>NNS</td>
<td>D</td>
<td>HEI Standards-2010</td>
<td>A</td>
<td>III</td>
<td>(3)(b), (3)(e)</td>
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<td>d. Non-safety-related piping and valves located at indoor</td>
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<td>NNS</td>
<td>D</td>
<td>ASME B31.1-2010</td>
<td>A</td>
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<td>a. DC equipment necessary for safety-related function</td>
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### d. Others

| 1) Non-safety-related battery chargers | TGB, CPB    | NNS          | N/A           | N/A                 | III                  |
| 2) Non-safety-related batteries       | TGB, CPB    | NNS          | N/A           | N/A                 | III                  |
| 3) Non-safety-related DC control centers | TGB, CPB    | NNS          | N/A           | N/A                 | III                  |

### 24. DE – Radioactive Drain

| a. Exposed piping and components in safety-related areas | AB         | NNS          | D             | ASME B31.1 - 2010 HI Standards - 2010 | A                     | II               | (3)(d)  |
| b. Embedded piping and components necessary for flood protection in safety-related areas | AB         | NNS          | D             | ASME B31.1 - 2010 HI Standards - 2010 | A                     | II               | (3)(d)  |
| c. Piping and components necessary for flood protection in non-safety-related areas | AB         | NNS          | D             | ASME B31.1 - 2010 HI Standards - 2010 | A                     | II               | (3)(d)  |
| d. Containment isolation valves and associated piping | RCB, AB     | SC-2         | B             | ASME Section III NC - 2007 with 2008 addenda | Yes                   | I                |
| e. Flood alarm loops of ESF pump rooms and elevation 55 ft 0 in of each quadrant wall | AB         | SC-3         | C             | ASME Section III ND - 2007 with 2008 addenda | Yes                   | I                |
| f. Reactor containment bldg. drain sump pump | RCB         | NNS          | D             | HI Standards - 2010     | A                     | II               | (3)(d)  |
### Table 3.2-1 (23 of 86)

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<td>h. Aux. bldg. floor drain sump pump</td>
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<td>j. Safety injection pump room sump pump</td>
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<td>l. Containment spray pump room sump pump</td>
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<td>d. Expansion tanks</td>
<td>AB, EDGB</td>
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<td>e. Combustion air intake duct work</td>
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<td>ASME AG-1 - 2009</td>
<td>Yes</td>
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<td>f. Engine, engine-mounted components, and generator</td>
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<td>G</td>
<td>IEEE Std. 387-1995</td>
<td>Yes</td>
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<td>g. Starting air compressors, air dryer package, lube oil separator, lube oil/preheating water heat exchanger, HT water electric heater, preheating HT water pump, prelube oil pump and other non-safety-related equipment</td>
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<td>i. Exhaust piping</td>
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<td>26. DI – Display</td>
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**28. DP – Diverse Protection System (DPS)**

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<th>AB</th>
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**29. ED – Non-radioactive equipment vent and drain**

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<th>a. Non-safety-related components and piping in safety-related areas</th>
<th>AB, CCWHXB, CPB, ESWB</th>
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<td>h. Valves and piping of makeup water supply line and boric acid makeup line from V1208, V1210, V2001 and V2002 to cooling loop</td>
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#### 37. FI – Fixed In-core Detector Amplifier System (FIDAS)

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#### 38. FP – Fire Protection

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#### 1) Subsystem Components

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#### 2) Subsystem fire protection piping

| AB | NNS | D | ASME B31.1-2010 | A | I | (3)(c) |

#### 3) Subsystem fire protection piping

| AB | NNS | E | NFPA 13-2013 | A | I | (3)(c) |

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<td>c. Air compressor and auxiliaries</td>
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<td>d.  IRWST sump strainers</td>
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<td>e.  Swing panels</td>
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#### 52. LD – Leak Detection

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<td>TGB CPB CW Pump House</td>
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<td>f. RCP vibration monitoring (RCPVMS)</td>
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#### 62. NP – 13.8 kV Non-1E Power

| a. 13.8 kV switchgear (SW02N) in TG building | TGB | NNS | N/A | N/A | A | III | (3)(e) |
| b. 13.8 kV switchgear (SW02M) in TG building | TGB | NNS | N/A | N/A | N/A | III |
| c. 13.8 kV switchgears in aux. building | AB | NNS | N/A | N/A | A | II | (3)(d) |

#### 63. NR – Ex-core Neutron Flux Monitoring

| a. Startup/control channel | RCB, AB | NNS | N/A | N/A | A | II | (3)(d) |

#### 64. PD – Diverse Indication System (DIS)

| a. DIS cabinet | AB | NNS | N/A | IEEE Std. 344-2004, IEEE Std. 384-1992 | A | II | (3)(d) |
| b. DIS display (FPD) & switch | AB | NNS | N/A | IEEE Std. 344-2004 | A | II | (3)(d) |

#### 65. PE – Engineered Safety Features-Component Control

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### b. Safety-Related Instrument Sensing Line

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### 66. PF – 4.16 kV Class 1E Power

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<td>b. Meeting room Workstation</td>
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#### Table 3.2-1 (47 of 86)

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| 7) Diverse Indication System FPD | AB | NNS | N/A | N/A | A | II | (3)(d) |
| 8) Mark-VI FPD | AB | NNS | N/A | N/A | A | II | (3)(d) |

9) ESCM  

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#### 70. PN – NSSS Process Instrumentation

a. Safety-related

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b. Non-safety-related

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#### 71. PO – Process Component Control

a. Non-safety-related component control cabinet and local installation component

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| 2) Reliability related process loop controller cabinet and related components | TB | NNS | N/A | IEEE Std. 383-2003(17) | A | II | (3)(d), (3)(e) |
|                                |             |              |                | IEEE Std. 420-2001(17) |                      |                   |         |
|                                |             |              |                | IEEE Std. 7-4.3.2-2003(17) |                    |                   |         |
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#### b. Non-safety-instrument sensing lines

| 1) Non-safety instrument sensing line in safety-related area | AB, RCB, TGB, CPB, ACC, GTGB, ESWB, CWPHT, CCWHXB, Yard | NNS | D | N/A | A | II | (3)(d) |
| 2) Reliability related non-safety instrument sensing line in non-safety-related area | AB, TGB, CPB, FPWTB, RCB, CWPHT, Yard | NNS | D | N/A | A | III | (3)(e) |
| 3) Other non-safety instrument sensing line in non-safety-related area | AB, TGB, CCWHXB, RCB, CPB, CWPHT, FPWTB, ESWB, AAC, GTGB, Yard | NNS | D | N/A | N/A | III | |

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<td>a. Instrumentation for Type A, B, and C variables</td>
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<td>c. Instrumentation for Type E variables</td>
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<td>2) Non-safety-related equipment in safety-related areas and TSC</td>
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#### 74. PS – Process Sampling

| a. Steam generator sample lines located in the reactor containment building including containment isolation | RCB, AB, CPB | SC-2 | B | ANSI/ISA-S67.02.01 | Yes | I |
| b. Non-safety-related sample lines in the safety-related areas | AB | NNS | D | N/A | A | II | (3)(d) |
| c. Equipment contacting with radioactive sample in non-safety-related areas | CPB | NNS | D | N/A | N/A | III |
| d. Analyzer and instrumentation equipment | AB, CPB, TGB | NNS | N/A | N/A | N/A | III |
| e. Others | AB, TGB | NNS | D | N/A | N/A | III |

#### 75. PW – Power Control System

| a. Reactor regulating system – signal conditional and processing electronics | AB | NNS | N/A | N/A | A | II | (3)(d) |
### Table 3.2-1 (51 of 86)

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<td><strong>d.</strong> Reactor power cutback system – signal conditional and processing electronics</td>
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<td>N/A</td>
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<td>(3)(d)</td>
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<td><strong>e.</strong> DRCS remote I/O cabinet</td>
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<td>N/A</td>
<td>N/A</td>
<td>I</td>
<td>(N-9)</td>
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76. **PX – Primary Sampling**

<p>| a. RCS hot leg sample to but excluding line CIV inside Containment | RCB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |
| b. RCS hot leg sample line from but excluding CIV outside Containment to post-accident primary sample cooler rack | AB | NNS | D | ASME B31.1-2010 | A | I |
| c. RCS PZR surge sample line to but excluding CIV inside containment | RCB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |
| d. RCS PZR Steam Space Sample Line to but excluding CIV inside Containment | RCB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |
| e. SI pumps miniflow sample line isolation valves | AB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |
| f. CS pump miniflow sample line isolation valves | AB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |
| g. SC pump miniflow sample line isolation valves | AB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |</p>
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<th>Seismic Category</th>
<th>Remarks</th>
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<td>h. SI tank sample line to but excluding CIV inside containment</td>
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<td>i. Containment air sample line to but excluding CIV inside containment</td>
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<td>j. Containment air sample return line from but excluding CIV inside containment</td>
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<td>NNS</td>
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<td>k. PASS sample return line from but excluding CIV inside containment</td>
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<td>l. CVCS purification filter and ion exchanger common outlet sample line isolation valves</td>
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<td>m. Sampling return line to VCT from but including VCT isolation valves</td>
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<td>o. Containment isolation valves and associated piping</td>
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77. QN – Qualified Indication and Alarm – Non-Safety (QIAS-N)

| a. QIAS-N display device (QIAS-N FPD, mini-LDP and SODP) | AB | NNS | N/A | IEEE Std. 384-1992 | A | I | (N-9) |
| b. QIAS-N processing device and related equipment | AB | NNS | N/A | IEEE Std. 384-1992 | A | I | (N-9) |

78. QP – Qualified Indication and Alarm – P(QIAS-P)

| a. QIAS-P display | SC-3 | N/A | IEEE Std. 323-2003 | Yes | I |
| c. HJTC instrumentation flange assembly | SC-1 | A | IEEE Std. 7-4.3.2-2003 | Yes | I |
| d. Heated junction thermocouple probe assembly | SC-1/SC-3 | A/C | IEEE Std. 7-4.3.2-2003 | Yes | I | SC-1: Seal plug |
### Table 3.2-1 (54 of 86)

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<td>ASME Section III, NF -2007 with 2008 addenda</td>
<td>Yes</td>
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<td>4) Upper air plenum</td>
<td>RCB</td>
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<td>ASME Section III, NF -2007 with 2008 addenda</td>
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<tr>
<td>i. Core support structures and Internal Structures</td>
<td>RCB</td>
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<td>Yes</td>
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<td>j. Valves</td>
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<td>1) Pressurizer spray control valves</td>
<td>RCB</td>
<td>SC-1</td>
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<td>2) Pressurizer spray isolation valves</td>
<td>RCB</td>
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<td>A</td>
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<td>Yes</td>
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<td>3) Downstream of flow restricting devices</td>
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<td>SC-2</td>
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<td>4) 3-way valves of POSRV discharge piping</td>
<td>RCB</td>
<td>SC-3</td>
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<td>k. Discharge piping vacuum breaker</td>
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<td>SC-3</td>
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<td>ASME Section III ND-2007 with 2008 addenda</td>
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<td>l. RCP lube oil collection tank</td>
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80. RG – Reactor Coolant Gas Vent

| | Location(2) | Safety Class | Quality Group | Codes and Standards | 10 CFR 50, App. B(3) | Seismic Category |
| a. Pressurizer gas vent piping upstream of and including the vent isolation valves V412 and 413 | RCB | SC-1 | A | ASME Section III NB-2007 with 2008 addenda | Yes | I |
| b. Reactor vessel upper head gas vent piping upstream of and including the vent isolation valves V416 and 417 | RCB | SC-1 | A | ASME Section III NB-2007 with 2008 addenda | Yes | I |
| c. RCGVS gas vent piping from downstream of the vent isolation valves V412,413,416, and 417 to and including the vent isolation valves V418,419, and 420 | RCB | SC-2 | B | ASME Section III NC-2007 with 2008 addenda | Yes | I |
### Table 3.2-1 (57 of 86)

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<th>SSC Identification</th>
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<td>d. RCGVS gas vent piping from downstream of the vent isolation valves V418 to RDT</td>
<td>RCB</td>
<td>NNS</td>
<td>D</td>
<td>ASME B 31.1-2010</td>
<td>A</td>
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<td>e. RCGVS gas vent piping from downstream of the vent isolation valves V419, 420 to the IRWST anchor wall</td>
<td>RCB</td>
<td>NNS</td>
<td>D</td>
<td>ASME B 31.1-2010</td>
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<td>f. RCGVS gas vent piping from downstream of the IRWST anchor wall to the end point of RCGVS sparger</td>
<td>RCB</td>
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<td>ASME Section III ND-2007 with 2008 addenda</td>
<td>Yes</td>
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</table>

81. RP – Reactor Protective

| | Location(2) | Safety Class | Quality Group | Codes and Standards | 10 CFR 50, App. B(3) | Seismic Category | Remarks |
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<th>Codes and Standards</th>
<th>10 CFR 50, App. B (^{(3)})</th>
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<td>82. RS – Remote Shutdown Room</td>
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<td>b. IFPD</td>
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<td>NNS</td>
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<td>N/A</td>
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<td>d. Shutdown overview display panel (SODP)</td>
<td>AB</td>
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<td>N/A</td>
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<td>f. Non-Class 1E switch</td>
<td>AB</td>
<td>NNS</td>
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<td>II</td>
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<td>83. RW – Radwaste Control Room</td>
<td>CB</td>
<td>NNS</td>
<td>N/A</td>
<td>N/A</td>
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<td>84. SA – Service Air</td>
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<td>b. Non-safety-related piping and equipment in safety-related areas</td>
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<td>c. Air compressor and auxiliaries</td>
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<td>e. Others</td>
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<td>1) SCS suction piping and valves on the RCS side from RCS hot leg up to including SI-653, 654</td>
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<td>SC-1</td>
<td>A</td>
<td>ASME Section III NB-2007 with 2008 addenda</td>
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<td>2) SC piping and valves from downstream of SI-653, 654 to the connecting point to SIS in the downstream of SI-601/600</td>
<td>RCB, AB</td>
<td>SC-2</td>
<td>B</td>
<td>ASME Section III NC-2007 with 2008 addenda</td>
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<td>3) Piping and valves on the IRWST cooling line from downstream of SI-688, 693 to SI-300, 301 (up to and including SI-391)</td>
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<td>5) Piping and valves on the SCS filling line from and including SI-708, 709 to upstream of SI-106</td>
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<td>6) Radioactive drain system connection piping</td>
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<td>II</td>
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<td>7) All relief valves discharge piping</td>
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<td>a. Valves and piping from SG up to the anchor wall of the blowdown flash tank room, including containment isolation valves</td>
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<td>SC-2</td>
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<td>ASME Section III NC-2007 with 2008 addenda</td>
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<td>c. Regenerative heat exchanger</td>
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<td>d. Mixed bed demineralizer</td>
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<td>e. Valves and piping from the anchor wall of the blowdown flash tank room to the point (V050) where discharged into the condensate</td>
<td>AB</td>
<td>NNS</td>
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<td>ASME B31.1-2010</td>
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<td>f. Valves and piping from the points (V1045, 050) where discharged into the condensate, and the wastewater treatment system to the auxiliary building wall.</td>
<td>AB</td>
<td>NNS</td>
<td>D</td>
<td>ASME B31.1-2010</td>
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<td>g. Valves and piping from the blowdown flash tank to MSVH</td>
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<td>i. Valves and piping except (e), (f), (g), and (h) within auxiliary building</td>
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<td>NNS</td>
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<td>ASME B31.1-2010</td>
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<td>II</td>
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<td>j. Valves and piping from the point (V050), where discharged into the condensate, to the liquid radwaste system</td>
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<td>ASME B31.1-2010</td>
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<td>k. Equipment and piping within the compound building and turbine building</td>
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<td>1. Wet lay-up recirculation pump</td>
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<td>A</td>
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<td>m. Valves and piping from wet lay-up subsystem within containment or auxiliary building except the containment penetration area and pressure boundaries.</td>
<td>RCB, AB</td>
<td>NNS</td>
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<td>n. Valves and piping of the containment penetration area and SG pressure boundaries</td>
<td>RCB, AB</td>
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<td>o. Steam generator blowdown prefilters and postfilter</td>
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### 87. SI – Safety Injection

<p>| a. Safety injection pumps | AB          | SC-2         | B             | ASME Section III NC-2007 with 2008 addenda | Yes                 | I                |         |
| b. Safety injection tanks | RCB         | SC-2         | B             | ASME Section III NC-2007 with 2008 addenda | Yes                 | I                |         |
| c. Safety injection filling tank | AB          | NNS          | D             | ASME Section VIII-2007 with 2008 addenda | N/A                 | III              |         |
| d. Piping and valves | | | | | |
| 1) SIP miniflow line (from SIP orifice or SI-218, 219, 254, 255 to IRWST) | AB          | SC-2         | B             | ASME Section III NC-2007 with 2008 addenda | Yes                 | I                |         |
| 2) SI piping and valves from IRWST to upstream of and excluding the check valves SI-543, 541, 542, 540 and SI-523, 533 | RCB, AB     | SC-2         | B             | ASME Section III NC-2007 with 2008 addenda | Yes                 | I                |         |</p>
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<th>Location(2)</th>
<th>Safety Class</th>
<th>Quality Group</th>
<th>Codes and Standards</th>
<th>10 CFR 50, App. B(3)</th>
<th>Seismic Category</th>
<th>Remarks</th>
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<tr>
<td>3) SI piping and valves between the DVI nozzle and including the check valves SI-543, 541, 542, 540, SIT check valve SI-245, 225, 235, 215, and SI-648, 628, 638, 618</td>
<td>RCB</td>
<td>SC-1</td>
<td>A</td>
<td>ASME Section III NB-2007 with 2008 addenda</td>
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<td>4) Hot leg injection piping downstream of and including SI-523, 533 and SI-322, SI-332 to the piping connected to the SCS suction line.</td>
<td>RCB</td>
<td>SC-1</td>
<td>A</td>
<td>ASME Section III NB-2007 with 2008 addenda</td>
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<td>5) Piping and valves on the SIT filling and drain line from and including SI-290 to up to and including SI-661 and up to and excluding SI-322, SI-332, SI-245, 225, 235, 215, SI-648, 628, 638, 618. Piping between the valves SI-290 and SI-293 is not included.</td>
<td>RCB, AB</td>
<td>SC-2</td>
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<td>6) SIT filling piping between the valves SI-290 and SI-293 (excluding SI-290 and SI-293)</td>
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<th>Seismic Category</th>
<th>Remarks</th>
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<td>7) SIT nitrogen supply piping up to and including valves SI-642, 622, 632, 612, 649, 629, 639, 619</td>
<td>RCB</td>
<td>SC-2</td>
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<td>10 CFR 50, App. B(3)</td>
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<td>9) Piping and valves on the SIS filling line from and including SI-700, 714, 701, 715 to the piping downstream of SI-476, 435, 478, 447</td>
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<td>SC-2</td>
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<td>ASME Section III NC-2007 with 2008 addenda</td>
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<td>10) Piping and valves on the SIS filling line from SIFT to up to and including SI-722 and up to and excluding SI-700, 714, 701, 715 and SI-708, 709</td>
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<td>NNS</td>
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<td>11) SIFT vent line</td>
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<td>12) Radioactive drain system connection piping</td>
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b. Compound building chilled water subsystem

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106.WM – Demineralized Water Makeup

a. Containment isolation valves and associated piping

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b. Non-safety-related components and piping in safety-related areas

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107. WO – Essential Chilled Water

| a. Chillers                                              | AB                  | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda               | Yes                         | I                |         |
| b. Pumps                                                 | AB                  | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda               | Yes                         | I                |         |
| c. Compression tanks and air separators                  | AB                  | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda               | Yes                         | I                |         |
| d. Control valves                                        | AB                  | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda               | Yes                         | I                |         |
| e. Manual valves                                         | AB                  | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda               | Yes                         | I                |         |
| f. Piping                                                | AB                  | SC-3         | C             | ASME Section III ND-2007 with 2008 addenda               | Yes                         | I                |         |
| g. Chemical additive tanks                               | AB                  | NNS          | D             | ASME Section VIII-2010                                   | A                           | II               | (3)(d)  |
| h. Refrigerant exhaust piping                            | AB                  | NNS          | D             | ASME Section VIII-2010                                   | A                           | II               | (3)(d)  |
| i. Demineralized water makeup control valves             | AB                  | NNS          | D             | ASME B31.1-2010                                          | A                           | II               | (3)(d)  |
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108.WT – Turbine Generator Building Closed Cooling Water

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109.WV – Liquid Radwaste

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<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API650 – 2007</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>k. Caustic batch tank</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API650 – 2007</td>
<td>N/A</td>
<td>III</td>
<td>Not containing radioactive material</td>
</tr>
<tr>
<td>l. Chemical additive tank</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API650 – 2007</td>
<td>N/A</td>
<td>III</td>
<td>Not containing radioactive material</td>
</tr>
<tr>
<td>m. Floor drain pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>n. Equipment waste pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>o. Chemical waste pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>p. Monitor tank pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>q. Seal water pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>r. Acid batch pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>N/A</td>
<td>III</td>
<td>Not containing radioactive material</td>
</tr>
</tbody>
</table>
# APR1400 DCD TIER 2

## Table 3.2-1 (83 of 86)

<table>
<thead>
<tr>
<th>SSC Identification</th>
<th>Location(2)</th>
<th>Safety Class</th>
<th>Quality Group</th>
<th>Codes and Standards</th>
<th>10 CFR 50, App. B (3)</th>
<th>Seismic Category</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>s. Caustic batch pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>N/A</td>
<td>III</td>
<td>Not containing radioactive material</td>
</tr>
<tr>
<td>t. Chemical additive pump</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>API610 - 2010</td>
<td>N/A</td>
<td>III</td>
<td>Not containing radioactive material</td>
</tr>
<tr>
<td>u. LRS seal water heat exchanger</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>ASME Section VIII - 2007 TEMA – 2007</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>v. RO package</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>ASME Section VIII – 2007</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
</tbody>
</table>

### 110. WX – Solid Radwaste

| a. Piping and components in safety-related areas | AB | NNS | D | ASME B31.1 - 2010 ASME Section VIII – 2007 | A | II | (3)(d) |
| b. Solid waste compactor | CPB | NNS | D | N/A | N/A | III |
| c. Piping and components containing radioactive material | AB, CPB | NNS | D | ASME B31.3 - 2010 ASME Section VIII - 2007 API-650 – 2007 | Note (4) | Note (4) |
| d. Low-activity spent resin tank | CPB | NNS | D | ASME Section VIII - 2007 | Note (4) | Note (4) |
| e. Spent resin long term storage tank | CPB | NNS | D | API650 – 2007 | Note (4) | Note (4) |
| f. New resin tank | AB | NNS | D | ASME Section VIII – 2007 | A | II | (3)(d) |

### 111. WY – Detergent Waste

| a. Detergent waste tank | CPB | NNS | D | API650 – 2007 | Note (4) | Note (4) |
| b. Detergent waste tank pump | CPB | NNS | D | API610 – 2010 | Note (4) | Note (4) |
### Table 3.2-1 (84 of 86)

<table>
<thead>
<tr>
<th>SSC Identification</th>
<th>Location</th>
<th>Safety Class</th>
<th>Quality Group</th>
<th>Codes and Standards</th>
<th>10 CFR 50, App. B(3)</th>
<th>Seismic Category</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>c. Detergent waste filter</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>ASME Section VIII - 2007</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
<tr>
<td>d. Piping and valves containing radioactive material</td>
<td>CPB</td>
<td>NNS</td>
<td>D</td>
<td>ASME B31.3-2010</td>
<td>Note (4)</td>
<td>Note (4)</td>
<td></td>
</tr>
</tbody>
</table>

112. DT - Turbine Generator Building Drain
The COL applicant is to provide the classification of structures, systems, and components for turbine generator building drain system (COL 3.2(3)).

(1) As used in this document, the term safety-related area applies to those areas containing equipment or structures required for safe shutdown (including accident mitigation).

(2) Locations are defined below:

<table>
<thead>
<tr>
<th>Code</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>RCB</td>
<td>Reactor Containment Building</td>
</tr>
<tr>
<td>CPB</td>
<td>Compound Building</td>
</tr>
<tr>
<td>CCWHXB</td>
<td>Component Cooling Water Heat Exchanger Building</td>
</tr>
<tr>
<td>CWP H</td>
<td>Circulating Water Pump House</td>
</tr>
<tr>
<td>MSVH</td>
<td>Main Steam Valve House</td>
</tr>
<tr>
<td>EOF</td>
<td>Emergency Operation Facility</td>
</tr>
<tr>
<td>AAC GTGB</td>
<td>Alternate Alternating Current Gas Turbine Generator Building</td>
</tr>
<tr>
<td>SWYD</td>
<td>Switchyard</td>
</tr>
<tr>
<td>AB</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>TGB</td>
<td>Turbine Generator Building</td>
</tr>
<tr>
<td>ESBW</td>
<td>Essential Service Water Building</td>
</tr>
<tr>
<td>FPWTB</td>
<td>Fire Pump &amp; Water/Wastewater Treatment Building</td>
</tr>
<tr>
<td>FHA</td>
<td>Fuel Handling Area</td>
</tr>
<tr>
<td>EDGB</td>
<td>Emergency Diesel Generator Building</td>
</tr>
</tbody>
</table>

(3) Legend:

- **Yes** – Compliance with the requirements of 10 CFR Part 50, Appendix B, is required.
- **A** – Augmented quality assurance requirements of Appendix B to 10 CFR Part 50, is applied. Augmented quality controls are applied to the following areas:
  - (a) ATWS (Anticipated Transient without Scram)
  - (b) Station Blackout
  - (c) Fire Protection
  - (d) Seismic Category II SSCs, external injection provision to cope with severe accident
  - (e) Risk significant non-safety-related SSCs determined by design RAP
- **N/A** – The requirements of 10 CFR Part 50, Appendix B, are not required.
Table 3.2-1 (85 of 86)

(4) Designed in accordance with NRC RG 1.143. The radwaste facilities, including the structures, systems, and components, are designed to meet the design basis loads, including the natural phenomena and internal and/or external man-induced hazards design criteria, in accordance with NRC RG 1.143.

- The radwaste safety classifications for the radioactive waste management systems: LWMS, GWMS, SWMS, and the SGBD systems and components, are presented in Sections 11.2, 11.3, 11.4, and 10.4.8, respectively.
- The radwaste safety classification for the compound building and the auxiliary building, in which the radioactive waste management systems (LWMS, GWMS, and SWMS) and the components for the SGBD system are housed respectively, is RW-IIa in accordance with the guidance in NRC RG 1.143.

(5) Designed based on guidance contained in NRC NUREG-0696 and NUREG-0737, Supplement 1.

(6) Security system requirements per 10 CFR Part 73.

(7) NRC RG 1.97 endorses IEEE Std. 497 as an acceptable method for providing instrumentation to monitor variables for accident conditions, subject to the 5 NRC positions. Instrumentation meets qualification and quality requirements of NRC RG 1.97 and IEEE Std. 497.

(8) Guidance per NUREG-0718 and NRC RG 1.47.

(9) Earthquake monitoring is per NRC RG 1.12.

(10) Design guidance per NRC RG 1.13.

(11) Design guidance per NRC RG 1.13, NUREG-0554, and NUREG-0612.

(12) Design guidance per NRC RG 1.189.

(13) The entire crane, including the bridge and trolley, is designed and constructed in accordance with NRC RG 1.29.

(14) Non-safety-related diverse protection system per 10 CFR 50.62 and GL 85-06.

(15) Non-safety-related ACUs and components, including fan/motor and associated isolation dampers, are designed and constructed per NRC RG 1.140.

(16) Design guidance per NRC RG 1.45.

(17) These codes and standards are applied to requirements of interface design.
NSSS Notes:

(N-1) Two safety classes are used for heat exchangers to distinguish primary and secondary sides where they are different.
(N-2) All of the reactor internals are classified as seismic Category I.
(N-3) The reactor coolant pump (RCP) auxiliaries are not required to be classified as seismic Category I because 1) continuous operation of the pumps is not required during or following an SSE and 2) sufficient time is available (i.e., >30 minutes) for the operator to stop the RCPs before loss of the RCP auxiliaries would cause any damage to the RCPs (see CENPD-201-A).
(N-4) Only those structural portions of the RCPs that comprise the RCPB and, therefore, support the safety function of the integrity of the RCPB must be categorized as Safety Class I.
(N-5) Safety classes of piping within the RCPB (as defined in 10 CFR 50) are selected in accordance with the ANSI/ANS 51.1 criteria. Safety Classes 1, 2, 3 and Non-Nuclear Safety of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C, and D of NRC RG 1.26, respectively.
(N-6) Flow-restricting orifices are provided in the nozzles for the RCS sampling lines, the pressurizer (PZR) level and pressure instruments, the RCP differential pressure instrument lines, the common SI header pressure instrument lines, the RCP seal pressure instrument lines, the charging line differential pressure instrument line, and the SI hot leg injection pressure instrument lines to limit flow in the event of a downstream break of a nozzle. The orifice size, 5.55 mm (7/32 in.) diameter x 25.4 mm (1 in.) long, precludes exceeding fuel design limits while using minimum makeup rates. This permits orderly shutdown in the event of a downstream break in accordance with 10 CFR Part 50, Appendix A, GDC 33. A reduction may therefore be made in the classification of downstream lines of the orifice.
(N-7) All containment isolation valves (and their operators) within NSSS scope of supply including manual valves, check valves, and relief valves, which also serve as isolation valves, are subject to the pertinent requirements of the Quality Assurance Program.
(N-8) The POSRVs are used for overpressure protection and rapid depressurization function.
(N-9) The “Associated Circuits” are defined, in accordance with IEEE Std. 384, as equipment, components, or systems the functions of which are Non-Nuclear Safety (NNS) and electrically Non-Class 1E, though their failures or abnormal states can affect the Class 1E equipment, components, or systems due to the effects of less than the minimum separation or the absence of electrical isolation from the Class 1E equipment, components, or systems. Consequently, the equipment, components, or systems, which are defined as “Associated Circuits” although they are functionally Non-Nuclear Safety, are subject to the qualification requirements placed on Class 1E equipment, components, or systems.
(N-10) Codes and standards are not specified because CEDM motor assembly and extension shaft assembly are neither pressure boundary components nor electric component. Safety function of those components is limited to scramability.
Table 3.2-2

Classification System Relationship

<table>
<thead>
<tr>
<th>Quality Group</th>
<th>Safety Class</th>
<th>Seismic Category</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>SC-1</td>
<td>I</td>
</tr>
<tr>
<td>B</td>
<td>SC-2</td>
<td>I</td>
</tr>
<tr>
<td>C, G</td>
<td>SC-3</td>
<td>I</td>
</tr>
<tr>
<td>D, E</td>
<td>NNS</td>
<td>I (1)</td>
</tr>
</tbody>
</table>

(1) Seismic Category I items that are not safety-related but are required by regulatory requirements to be designed for seismic loads are Quality Group D or E (e.g., portions of fire protection system).
3.3 Wind and Tornado Loadings

All seismic Category I and II structures, systems, and components (SSCs), except those not exposed to wind, are designed for wind and tornado/hurricane loadings.

3.3.1 Wind Loadings

The design wind loadings on the surfaces of seismic Category I and II SSCs subject to wind are determined in accordance with ASCE/SEI 7-05 (Reference 1).

Load combinations including wind loads are discussed in Section 3.8.

3.3.1.1 Design Wind Velocity and Recurrence Interval

The 50-year 3-second wind gust speed of 64.8 m/s (145 mph) corresponds to the wind speed at 10 m (33 ft) above ground for exposure Category C (open terrain), as defined in Section 6.5.6.3 of ASCE/SEI 7-05 (Reference 1). A recurrence interval of the design wind speed is 50 years with a 0.02 annual probability of being exceeded.

The design wind speed specified for the APR1400 standard plant covers most of the contiguous United States based on the basic wind speed map provided in Figure 6-1 of ASCE/SEI 7-05 (Reference 1).

The COL applicant is to demonstrate that the site-specific design wind speed is bounded by the design wind speed of 64.8 m/s (145 mph) (COL 3.3(1)).

3.3.1.2 Determination of Applied Forces

Wind pressure on the surfaces of the seismic Category I and II SSCs is determined in accordance with Equation (6-15) of ASCE/SEI 7-05 (Reference 1). The wind directionality factor ($K_d$), velocity pressure exposure coefficient ($K_z$), and topographic factor ($K_{zt}$) are defined in SRP 3.3.1 and conform with the provisions of SRP 3.3.1. For seismic Category I and II SSCs, an importance factor (I) of 1.15 is used in order to be compatible with the recurrence interval of 100 years. Exposure category D is applied in determining wind loads regardless of the site location.

Wind forces on the surfaces of seismic Category I and II structures, which are classified as enclosed or partially enclosed structures, are determined in accordance with the equation (6-17) of ASCE/SEI 7-05 (Reference 1).
All non-safety-related facilities subject to winds are designed using the same design wind speed and methodology applied to seismic Category I SSCs except the load combinations. A wind-loading-caused full or partial failure of seismic Category II SSCs adjacent to seismic Category I SSCs does not affect the ability of the seismic Category I SSCs to perform their intended safety functions. Otherwise, the seismic Category I SSCs are designed to maintain their integrity from the failure of seismic Category II SSCs.

The COL applicant is to demonstrate that the site-specific seismic Category II structures adjacent to the seismic Category I structures are designed to meet the provisions described above (COL 3.3(2)).

3.3.2 Tornado Loadings

The APR1400 standard and site-specific plant is designed to protect SSCs listed in the Appendix to NRC Regulatory Guide (RG) 1.117 (Reference 2) from tornadoes and hurricanes. All seismic Category I and II SSCs subject to tornado and hurricane winds are designed to meet the acceptance criteria described in Section 3.8.

3.3.2.1 Applicable Design Parameters

As provided in Table 2.0-1, the design basis tornado parameters are the same as those of Region I and categorized in NRC RG 1.76 (Reference 3). The annual probability of exceedance of the design basis tornado described above is $10^{-7}$, and the corresponding recurrence interval is approximately 10 million years.

The maximum wind speed of design basis hurricane is 116 m/s (260 mph) from the wind speed contour maps for hurricane-prone regions of the contiguous United States presented in NRC RG 1.221 (Reference 4). The annual probability of exceedance of the design basis hurricane is $10^{-7}$. The wind speed is nominal 3-second peak gust at a height of 10 m (33 ft) in flat open terrain.

The seismic Category I and II SSCs subject to extreme winds such as the design basis tornado or design basis hurricane are designed individually for the postulated extreme winds enveloping both the design basis tornado and design basis hurricane in terms of straight winds and wind-borne missiles. The pressure drop effects due only to the design basis tornado are combined with the design basis hurricane loadings, as described in Subsection 3.3.2.2.4.
The COL applicant is to perform an analysis if the site-specific wind and tornado/hurricane characteristics are not bounded by the site parameter postulated for the certified design (COL 3.3(3)).

3.3.2.2 Determination of Forces on Structures

The forces on seismic Category I and II SSCs due to the postulated extreme winds are obtained using methods outlined in Subsection 3.3.1.2. The missile barriers of the seismic Category I structures are designed based on the missiles listed in Table 3.5-2. The design method of missile barriers is presented in Subsection 3.5.3. The pressure drop effects due to the design basis tornado are determined using the guidance provided by Simiu and Scanlan (Reference 5). The loading combinations associated with the postulated extreme wind loadings are described in Tables 3.8-2, 3.8-7A, and 3.8-7B.

3.3.2.2.1 Hurricane Velocity Forces

Velocity forces due to the postulated extreme winds are determined using the approach described in Subsection 3.3.1.2 in conjunction with an importance factor (I) of 1.15 in accordance with SRP 3.3.2.

3.3.2.2.2 Hurricane Missile Effects

The missile barriers of seismic Category I structures are designed in accordance with the missile spectrum identified in Table 3.5-2. The missile barriers are designed to prevent the penetration, perforation, and withstand scabbing effects due to the hurricane missiles, as described in Subsection 3.5.3.

3.3.2.2.3 Tornado Pressure Drops

Pressure drop effects during the design basis tornado are evaluated based on the enclosure category of seismic Category I and II SSCs, as applicable. Vented or partially enclosed and enclosed buildings are designed to withstand the pressure drop while pressure drop effects are not considered in the interior of unvented structures.

3.3.2.2.4 Combined Extreme Wind Effects

The loading combinations of the individual extreme wind loading components are in accordance with SRP 3.3.2. The total extreme wind load W, used in the load combinations
described in Section 3.8 is determined for the combined effects using the following relationships:

\[
W_t = W_w \\
W_t = W_p \\
W_t = W_m \\
W_t = W_w + 0.5 W_p \\
W_t = W_w + W_m \\
W_t = W_w + 0.5 W_p + W_m 
\]

Where:

\[
W_t = \text{total extreme wind load} \\
W_w = \text{load from hurricane wind effect} \\
W_p = \text{load from tornado atmospheric pressure change effect} \\
W_m = \text{load from hurricane missile impact effect} 
\]

Pressure drop effects due only to the design basis tornado are combined with the design basis hurricane loadings.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Extreme Wind Loads

Failure of any SSCs not designed for postulated extreme wind loads does not affect the capability of safety-related SSCs to perform their intended safety functions.

The non-safety-related SSCs not designed for extreme wind loads are evaluated and designed using one of the following methods:

a. Designing the SSCs adjacent to seismic Category I SSCs to wind, and tornado/hurricane loadings
b. Investigating the effect of adjacent structural failure on seismic Category I SSCs to provide reasonable assurance that the ability of the seismic Category I SSCs to perform their intended safety functions is not impacted or affected

c. Designing and providing a structural barrier to protect seismic Category I SSCs from adjacent structural failure

The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for the extreme wind loads do not impact either the function or integrity of adjacent seismic Category I SSCs (COL 3.3(4)).

3.3.3 Combined License Information

COL 3.3(1) The COL applicant is to demonstrate that the site-specific design wind speed is bounded by the design wind speed of 64.8 m/s (145 mph).

COL 3.3(2) The COL applicant is to demonstrate that the site-specific seismic Category II structures adjacent to the seismic Category I structures are designed to meet the provisions described in Subsection 3.3.1.2.

COL 3.3(3) The COL applicant is to perform an analysis if the site-specific wind and tornado/hurricane characteristics are not bounded by the site parameter postulated for the certified design.

COL 3.3(4) The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for the extreme wind loads do not impact either the function or integrity of adjacent seismic Category I SSCs.

3.3.4 References


3.4 Water Level (Flood) Design

All seismic Category I structures, systems, and components (SSCs) are designed to withstand the effects of flooding due to natural phenomena or onsite equipment failures without loss of the capability to perform their safety-related functions.

The potential causes of external flooding include probable maximum precipitation, potential dam failures, and high groundwater and outdoor tank failures, and extreme sea waves such as storm surges, seiches, tsunamis, high tides, etc., as described in Section 2.4.

This analysis includes a site description and elevations of safety-related structures and equipment; evaluations of penetrations in seismic Category I structures; and the effects of flooding due to postulated pipe failures, inadvertent operation of fire protection systems, and failures of non-seismic and non-high-wind (including tornado and hurricane) protected tanks, vessels, and piping.

3.4.1 Flood Protection and Evaluation

3.4.1.1 Design Bases

The design basis flood level at the reactor site will be determined in accordance with NRC RG 1.59 (Reference 1) and ANSI/ANS 2.8 (Reference 2). Because the design basis flood level of the APR1400 standard design is at least 0.3 m (1 ft) below the plant grade as specified in Table 2.0-1, all safety-related SSCs located on the dry site as defined in NRC RG 1.102 (Reference 3) are protected from an external flood event.

The COL applicant is to provide the site-specific design of plant grading and drainage (COL 3.4(1)).

The COL applicant is to provide site-specific information on protection measures for the design basis flood, such levees, seawalls, flood walls, revetments or breakwaters or site bulkheads pursuant to NRC RG 1.102 as described in Subsection 2.4.10 (COL 3.4(2)).

All seismic Category I structures are designed to withstand the static and dynamic forces due to the maximum groundwater level, which is 0.61 m (2 ft) below the plant grade as provided in Table 2.0-1.
3.4.1.2 Flood Protection from External Sources

The flood protection measures for seismic Category I SSCs are designed in accordance with NRC RG 1.102 (Reference 3).

Seismic Category I structures identified in Table 3.2-1 are designed for flood protection. Seismic Category I structures are designed to protect safety-related equipment from flooding by incorporating the following safeguards into their construction:

a. No exterior access openings are lower than 0.41 m (1 ft 4 in.) above plant grade (yard grade) elevation.

b. The finished yard grade adjacent to the safety-related structures is maintained at least 0.41 m (1 ft 4 in.) below the ground floor elevation, except where ramps or steps are provided for access.

c. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood-level elevation.

d. Water seals are provided for all penetrations in exterior walls up to flood-level elevation. The water seals are designed for the static pressure of water at the flood elevation. Water seals in safety-related structures are designed to maintain integrity in the event of an SSE.

e. All below-grade exterior walls and basemats of seismic Category I structures are thickened by more than or equal to 0.6 m (2 ft) to protect against water seepage, as required in SRP 14.3.2.

Penetrations below the external flood level in the external walls of the auxiliary building include component cooling water, radwaste, and diesel fuel oil system piping and cable penetrations. Additional penetrations may be identified when layouts are finalized for systems such as sewage, demineralized water, station air, and security. All penetrations are sealed on the inside of the penetration to eliminate the potential of flooding through the penetration. The penetration seals are periodically inspected to ensure their functionality.
3.4.1.3 Flood Protection from Internal Sources

The APR1400 arrangement provides physical separation of redundant safety-related SSCs. The flood protection mechanisms related to minimizing the consequences of internal flooding include the following:

a. Structural enclosures or barrier walls

b. Drainage systems

c. Emergency sump

d. Internal curbs or ramps

e. Watertight doors

The APR1400 minimizes penetrations through enclosures or barrier walls below the flood level. Enclosures and barrier walls below the flood level are sealed to maintain watertightness. Barrier walls, floors, and penetrations are designed to withstand the maximum anticipated hydrodynamic loads associated with a pipe failure, as described in Section 3.6.

Divisional and quadrant separation by flood barriers with watertightness is provided for internal flood protection. Each quadrant is protected against propagation of internal flood event from one quadrant to any other.

The floor drainage systems are separated by quadrants with no common drain lines between the quadrants. Floors are gently sloped to allow for good drainage to the quadrant sumps. The functional capability of equipment and floor drainage system is maintained because the piping of the drainage system is designed as seismic Category II and embedded into the concrete of seismic Category I structures such as reactor containment building and auxiliary building.

The COL applicant is to establish procedures and programmatic controls to ensure the availability of the floor drainage (COL 3.4(3)).

The vertical and horizontal Emergency Overflow Lines (EOLs) are used to provide flow paths for the draining of flooded water, in addition to floor drains, as one of the flood mitigation measures. EOLs are the embedded pipes that connect rooms. EOLs are
seismically designed as seismic Category II. The sizes of the EOLs are determined based on the combination of available number of floor drains and required flow area needed to support drainage of the flooded water volume.

The lowest spaces of each building are designed as an emergency sump to keep flood water within the building where a flooding event could occur. The emergency sump is large enough to accommodate the volume of limiting flooding source.

Additionally, curbs or ramps and sealed penetrations function as flood barriers. Safety-related equipment and components are located at higher elevations so flooding events do not affect them.

Watertight doors are used for internal flood protection. Watertight doors are specified to withstand the static pressure from the maximum flood elevation as determined in the flooding analysis. Sensor signals of sensors to indicate the status of open and close of the watertight doors are provided to the main control room. The COL applicant is to periodically inspect watertight doors and the penetration seals to ensure their functionality (COL 3.4(4)).

The areas of concern in APR1400 are as follows:

a. Reactor containment building

The reactor containment building systems to be protected from flooding are the reactor coolant system (RCS), safety injection system (SIS), reactor coolant gas vent system (RCGVS), feedwater system (FWS), auxiliary feedwater system (AFWS), shutdown cooling system (SCS), component cooling water system (CCWS), and main steam system (MSS). The components to be protected from flooding are the valves and electric instrumentation of these systems.

b. Auxiliary building

The auxiliary building systems to be protected from flooding are the SIS, shutdown cooling system (SCS), chemical and volume control system (CVCS), containment spray system (CSS), auxiliary feedwater system (AFWS), and component cooling water system (CCWS). The components to be protected from flooding are the motor-driven pumps, valves, electrical equipment and instruments, Class 1E electric/instrumentation components, and cubicle coolers in the relevant system.
c. Emergency diesel generator building

The systems in the emergency diesel generator building to be protected from flooding are Class 1E emergency diesel generator system, and the emergency diesel generator fuel oil storage and transfer system. The components to be protected from flooding are diesel generator, diesel fuel oil transfer pump, and exhaust fan.

d. Site-specific safety structures

The COL applicant is to provide flooding analysis with flood protection and mitigation features from internal flooding for the CCW Heat Exchanger Building and ESW Building (COL 3.4(5)).

Tables 3.4-1 and 3.4-2 provide the locations of safety-related SSCs and a comparison of the maximum internal flood elevation in the vicinity of the components. Figures 3.4-1 through 3.4-7 provide the locations of watertight doors and flood barriers in the auxiliary building.

3.4.1.4 Evaluation of External Flooding

External flooding is evaluated based on flooding sources such as natural phenomena and the failure of onsite tanks or large buried pipes. The failure of non-safety-related onsite tanks such as condensate storage facilities (CSF) could result in a potential flood source. However, onsite tanks are located in the tank yard that is an adequate distance from safety-related structures, and watertight doors are installed at the exterior entrances located on the ground level of safety-related structures to prevent inflow of external flooding. Therefore, a non-safety-related tank failure does not result in adverse effects to safety-related SSCs. The maximum water level and flow velocity of an individual flood event are determined to estimate flood loads on seismic Category I structures and the watertightness of the structures during an external flood event. Seismic Category I structures are designed for the design basis flood level and the maximum groundwater level defined in Table 2.0-1.

The COL applicant is to provide the site-specific flooding hazards from engineered features, such as water tank collapsing, water piping breaking, etc. (COL 3.4(6)).
The COL applicant is to confirm that the potential site-specific external flooding events are bounded by design basis flood values or otherwise demonstrate that the design is acceptable (COL 3.4(7)).

No permanent dewatering systems are necessary to maintain safe and acceptable groundwater levels.

The COL applicant is to provide the site-specific dewatering system if the plant is built below the design basis flood level (COL 3.4(8)).

3.4.1.5 Evaluation of Internal Flooding

The internal flooding analysis demonstrates that plant nuclear safety functions are protected from the effects of internal flooding that are the result of a postulated failure or operation of the plant fire protection system. The safety-related SSCs included in Table 3.2-1 are protected against an internal flood. Potential flooding sources are as follows:

a. High- and moderate-energy piping failures

b. Full-circumferential ruptures in non-seismic moderate-energy piping

c. Postulated failures of non-seismic and non-tornado/hurricane-protected tanks and vessels

d. Pump mechanical seal failures

e. Operation of the fire protection system

Criteria and assumptions described in Subsection 3.6.2 are used for the internal flooding analysis. Subsection 3.6.2 provides the criteria used to define break and crack locations and configurations for high- and moderate-energy piping failures.

For flooding analysis, the single worst-case piping rupture for non-seismically analyzed piping is assumed for each analyzed area. Also, only one break at a time is postulated for non-seismic Category I or II piping as the result of a seismic event in the internal flooding analysis. All piping inside the reactor containment building and auxiliary building are seismically designed. Therefore, no breaks are postulated during a seismic event in these buildings. The discharge volume through the ruptured area is calculated in accordance with the formula given in ANSI/ANS 56.10, Section 3 (Reference 4). The released steam...
flow rate is conservatively assumed to be completely condensed to result in a higher flood level.

The discharge flow rate from a high-energy line break is obtained by one of the following critical flow correlations.

a. Moody model for two-phase mixture and saturated steam conditions

b. Henry-Fauske model for subcooled liquid

A LOCA that results in the largest discharge volume to the reactor containment is assumed as a flooding source in the flooding analysis of reactor containment building. The flood level of the reactor containment building is determined by dividing the accumulated volume by the total floodable area at El. 100 ft 0 in. For conservatism, the entire released volume is taken to accumulate at El. 100 ft of containment with no drainage to the HVT at a lower elevation before recirculation mode, although fluid flow to the lower elevation will actually be established when broken flow arrives at the openings to the lower elevation.

In the long term cooling period, during recirculation mode with containment spray from IRWST, the inflow of water onto El. 100 ft of containment consists of ECCS (SI and CS) injection flow from the IRWST with an outflow to the HVT and discharging back into the IRWST through spillways. The outflow to the IRWST through spillways is determined by using the general equation for pressure drop, Darcy’s formula.

Fire hose stations that could reach the area or zone where a fire occurs are contributed to internal flooding sources when a fire occurs. The discharge flow rate from fire hose stations is assumed to be 0.044 m$^3$/s (700 gpm).

The lowest level of the auxiliary building is designed to function as an emergency sump to collect flooding sources when a flood event occurs. The flood level of the emergency sump is determined by dividing the maximum volume of flooding sources by the floodable area of the emergency sump. The flood level, except for the lowest elevation, is determined based on the level established by the difference between the inflow rate of the postulated flooding source and the outflow rate through drains or openings in steady-state condition.

Fluid flow rates through stairwells, floor openings, and under door gaps are determined in accordance with the formulae given in ANSI 56.11 (Reference 5). The fluid flow rate through a stairwell or a floor opening is calculated using equation 5.2-1, and the flow rate
under a door is calculated using equation 5.2-3 in Reference 5. It is assumed that the total inventory of the IRWST and SI tanks are spilled out. The total water volume in the IRWST is considered as a limiting flood source in auxiliary building only as a result of a pipe rupture. No credit is taken for operation of sump pumps to mitigate the flooding consequences.

The internal flooding analysis is performed on a floor-by-floor and room-by-room basis.

Flooding analysis consists of the following steps:

a. Identification of safety-related SSCs
b. Identification of potential flooding sources
c. Determination of flow rates and flood levels
d. Risk assessment for components affected by a flood event
e. Determination of the need for protection and mitigation measures

3.4.1.5.1 Reactor Containment Building

The APR1400 is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCA's.

The reactor containment building is not designed to provide divisional separation but it allows flooding sources to flow to the lowest level of the building i.e., containment annulus area at El. 100 ft 0 in. through the floor openings and stairwells.

a. Water at El. 156 ft 0 in. flows to the lower elevation through four reactor containment fan cooler duct openings, two stairwells, and four safety injection tank openings.

b. Water at El. 136 ft 6 in. flows to the lower elevation through the grating on this floor. Water in the pressurizer cavity flows to the containment annulus area through the wire mesh door.
c. Water at El. 114 ft 0 in. flows to the lower elevation through the grating on this floor. Water in each valve room flows to the containment annulus area through the wire mesh door.

d. Water collected at El. 100 ft 0 in. flows to the holdup volume tank through the floor opening and then to the IRWST through the spillway. Water in the letdown heat exchanger room and reactor drain tank room flows to the containment annulus area through the wire mesh door.

The worst-case flooding event is a LOCA in reactor containment building because it results in maximum flood water volume to the reactor containment building as a flooding source. The water volume of bounding case is 109,197 ft³ which is summation of total discharged water volume from a LOCA (19,482 ft³) and the recirculation mode with IRWST water volume (89,715 ft³). There are flooding sources inside the reactor containment building such as reactor coolant system, main feedwater, auxiliary feedwater, component cooling water, fire protection and chemical and volume control systems.

The flooding source of an AF tank does not gravity discharge into the reactor containment building since the location of the piping is higher than the top level of the AF Tank.

The discharged water volume from the FW system during the isolation time by safety-related main feedwater isolation valve and the volume of AF makeup flow to steam generator are less than the volume of water from a LOCA in recirculation mode. For the FW line break evaluation, the transient water level is evaluated using a time history approach to determine the flood height and takes a credit of the flow path from the floor opening at El. 100 ft of RCB. Based on the sensitivity analysis, the flood level resulting from a postulated FW line break remains below El. 102 ft.

The water volume of FP system is isolated with flood protection measures including operator action. It has sufficient time for operators to identify and isolate the fire protection system in a flooding event. Non-seismically designed pipes are excluded as flood sources because all piping inside the reactor containment building is seismically designed. Operator action is credited for identification and isolation of flood sources that could result in larger volumes to containment than a worst case LOCA.

Discharged water first fills up the volume below elevation El. 100 ft 0 in. and then spreads the volume above the grade level of the reactor containment building. Water released by a
LOCA is collected in the IRWST through the spillways between HVT and IRWST. It then flows back to the reactor coolant system or is sprayed into the containment and recirculated.

The total discharged volume of a LOCA is 551.67 m$^3$ (19,482 ft$^3$) which consists of volume for reactor coolant system and four safety injection tanks described in Table 6.2.1-20 and Table 6.2.1-21. The net floodable volume under El. 100 ft 0 in., including volume of air space of IRWST 753 m$^3$ (26,592 ft$^3$), holdup volume tank 242.3 m$^3$ (8,557 ft$^3$), and normal sump 6.7 m$^3$ (237 ft$^3$) is 1,002 m$^3$ (35,385 ft$^3$). The total discharged water volume due to LOCA is smaller than the total floodable volume.

The flood level of the bottom area of the reactor containment building with total water volume of LOCA is determined to be 0.61 m (2 ft) from the El. 100 ft 0 in. The maximum containment flood level is determined by dividing total LOCA volume by floodable area 11,215 ft$^2$ of El. 100 ft. For LOCA evaluation, the analysis conservatively assumes that the break volume is discharged to containment with no loss of volume through the floor openings at El. 100 ft of the RCB prior to recirculation. The resulting containment flood level has been determined to be an additional 0.61 m (2 ft) to El. 102 ft. In the long term period with the containment water level at El. 102 ft, the outflow into the IRWST through two 24 inch spillways is greater than the inflow from SI and CS pumps into containment. The flood height decreases during the recirculation period, and, therefore, the flood water level remains less than the determined level of 0.61 m (2 ft) established before recirculation period. It envelops all flood levels throughout the entire containment area at El. 100 ft 0 in. The flood levels in the separately compartmentalized areas located above the bottom and annulus area are independently determined by taking account of flows in and out of these areas. The maximum flood level of containment does not affect safety-related equipment. There are no submerged SSCs required for safe shutdown. Table 3.4-1 provides a list and the locations of SSCs inside the reactor containment building that require flood protection. These SSCs are located above the maximum internal flood level.

**3.4.1.5.2 Auxiliary Building**

The auxiliary building is designed to provide physical separation to prevent spreading of fluids to the areas housing safety-related equipment and components.

**Elevation 55 ft 0 in**

The primary means of flood protection is the divisional or quadrant walls, which serve as flood barriers between redundant trains of safe shutdown systems and components. Flood
barriers provide separation between the quadrants, while maintaining equipment removal capability.

On the divisional wall, penetrations are sealed and no doors are provided up to El. 64 ft 0 in., which is the potential flood level from the bottom elevation. The heights of the divisional walls at El. 55 ft 0 in. are 13 ft (El. 68 ft 0 in.) as shown in Figures 1.2-11 and 1.2-12. The divisional walls are sufficiently high to contain total water volume in the affected quadrant. Watertight doors are provided between the quadrants to prevent potential flooding sources from spreading to adjacent quadrants.

The equipment to be protected at El. 55 ft 0 in. includes SI pumps, SC/CS pumps, heat exchangers, and CCW pumps. In each quadrant, the SI pump rooms, SC/CS pump rooms, and CCW pump rooms are separated by a flood barrier.

The following potential flooding sources are considered:

a. A postulated pipe failure is considered in only one area of the quadrant. During normal operation, a 0.25 m (10 in.) pipe crack in the SIS suction line from the IRWST is considered a potential flooding source.

b. There is no break in non-seismic moderate-energy piping because piping in the auxiliary building is designed as seismic Category I or II.

c. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate for firefighting equipment is assumed to be 0.044 m$^3$/s (700 gpm).

The worst case of flooding in the auxiliary building is the water source in the IRWST. The total water volume of the IRWST is 2,540 m$^3$ (89,715 ft$^3$) and the floodable area in the four quadrants (A, B, C, and D) is 1,168 m$^2$ (12,577 ft$^2$), 1,176 m$^2$ (12,664 ft$^2$), 1,232 m$^2$ (13,263 ft$^2$), and 1,232 m$^2$ (13,263 ft$^2$), respectively.

Based on these values, the maximum water level is 2.74 m (9 ft) with some margin. The released water volume is contained within the affected quadrant.

Elevation 78 ft 0 in

Flood water above El. 78 ft 0 in. drains to the lower elevation through the floor drain, stairwells, and openings. To avoid flooding adjacent quadrants, a curb or ramp is installed.
at each quadrant intersection. The emergency diesel generator is separated by distance and flood barriers. Radiation-control areas are also separated by flood barriers. A watertight door is provided to prevent spreading of flooding water through the potential flow path from the boric acid storage tank (BAST) tunnel to the auxiliary building.

The equipment to be protected from flooding at El. 78 ft 0 in. includes motor-driven AF pumps, turbine-driven AF pumps, Class 1E switchgear, essential chiller, and related electrical equipment. The turbine-driven AF pump and motor-driven AF pump are separated by a quadrant flood barrier.

The following potential flooding sources are considered:

a. A postulated pipe failure of a moderate-energy line is considered.

b. A high-energy line break event is not considered because there is no piping break in this area.

c. A break in non-seismic moderate-energy piping break is not considered because piping in the auxiliary building is designed as seismic Category I or II.

d. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate for firefighting equipment is assumed to be 0.044 m³/s (700 gpm).

The worst-case flooding scenario is a postulated pipe failure of a 0.10 m (4 in.) AF moderate-energy line in normal operation, during which flood water would drain to lower elevations through the drain system and openings. The potential flood level at this elevation is assumed as 0.15 m (6 in.).

AF pumps are located above the potential internal flood levels. Pressure door protection is provided for the turbine-driven auxiliary feedwater pump room to further protect these pumps and the adjacent motor-driven pumps from the impact of pipe ruptures. Electrical equipment is installed above the flood level so that flooding events do not affect this equipment. Therefore, AF pumps and electrical equipment are not flooded.

**Elevation 100 ft 0 in**

Flood water above El. 100 ft 0 in. drains to the lower elevation through floor drains, stairwells, and openings. To avoid flooding adjacent quadrants, a curb or ramp is installed
at each quadrant intersection. The emergency diesel generator is separated by distance and protected by flood barriers. The radiation control area is also separated by flood barriers. Watertight doors are installed to prevent spreading of flood water through the potential flow path from the compound building to the auxiliary building.

The equipment to be protected from flooding at El. 100 ft 0 in includes the 480V Class 1E motor control center, electrical equipment, and related cubicle coolers.

The following potential flooding sources are considered:

a. Postulated pipe failure of a moderate-energy line is considered.

b. A high-energy line break event is not considered because there is no piping break in this area.

c. There is no break of non-seismic moderate-energy piping because piping in auxiliary building is designed as seismic Category I or II.

d. The total water inventory of the volume control tank is considered a flooding source at quadrant A.

e. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate for firefighting equipment is assumed to be 0.044 m$^3$/s (700 gpm).

Based on flooding sources, the worst-case flooding scenario is postulated failure of a 0.36 m (14 in.) component cooling line in the general access area, during which flood water would drain to lower elevations through the drain system and openings. The potential flood level at this elevation is assumed as 0.15 m (6 in.).

The safety-related equipment and components are elevated above the flood level so that flooding events do not affect components. Therefore, the electrical equipment and related cubicle cooler are not flooded.

**Elevation 120 ft 0 in**

Flood water above El. 120 ft 0 in. drains to the lower elevation through floor drains, stairwells, and openings. To avoid flooding adjacent quadrants, a curb or ramp is installed at each quadrant intersection. The emergency diesel generator is separated by distance
and protected by flood barriers. The radiation control area is also separated by flood barriers.

The equipment to be protected from flooding at El. 120 ft 0 in includes the Class 1E motor control center, related cubicle coolers, safety injection containment isolation valves, and AF modulating valves.

The following potential flooding sources are considered:

a. A postulated pipe failure of a moderate-energy line is considered.

b. A high-energy line break is considered for the 0.10 m (4 in.) steam generator blowdown system line.

c. There is no break of non-seismic moderate-energy piping because piping in the auxiliary building is designed as seismic Category I or II.

d. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate for firefighting equipment is assumed to be 0.044 m³/s (700 gpm).

Based on flooding sources, the worst-case flooding scenario is the HELB of a 0.10 m (4 in.) steam generator blowdown system line. The floor drains are conservatively assumed to be clogged due to debris resulting from dynamic forces in the high-energy line break areas. The flood relief opening is installed toward the outside to remove flood water. A watertight door is installed to protect against spreading flood water to adjacent areas in the steam generator blowdown regenerator heat exchanger room.

In other areas except the SGBD regenerator heat exchanger room, the 0.10 m (4 in.) fire protection system line is considered a flooding source. The flood water is drained to lower elevations through the drain system and openings. The potential flood level at this elevation is assumed as 0.15 m (6 in.).

Safety-related equipment and components are elevated above flood level. Therefore, the Class 1E motor control center, cubicle coolers, and valves are not flooded.
Elevation 137 ft 6 in

Flood water above El. 137 ft 6 in. drains to the lower elevation through floor drains, stairwells, and openings. To avoid flooding adjacent quadrants, a curb or ramp is installed at each quadrant intersection. The remote shutdown room and reactor trip switchgear room are protected by flood barriers.

The equipment to be protected from flooding at El. 137 ft 6 in. includes Class 1E motor control center, switchgear, remote shutdown panel, main feedwater isolation valves, and main steam safety valves.

The following potential flooding sources are considered:

a. A postulated pipe failure of moderate-energy line is considered.

b. A high-energy line break is considered in the main steam enclosure and main steam valve room.

c. There is no break of non-seismic moderate-energy piping because piping in the auxiliary building is designed as seismic Category I or II.

d. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate for firefighting equipment is assumed to be 0.044 m³/s (700 gpm). The malfunction of sprinklers is also considered a potential flooding source.

A rupture of a feedwater system line is the worst case of flooding for the main steam valve room. The cross-section area of the break is based on 0.09 m² (1.0 ft²), as defined in Standard Review Plan, Branch Technical Position 3-3 (Reference 6). In addition, a main feedwater pump is assumed to operate at the maximum flow rate. The floor drains are conservatively assumed to be clogged due to debris resulting from dynamic forces in the high-energy line break areas. An emergency flood relief path is installed to drain out at each room. The potential flood level is 1.82 m (6 ft) above El. 137 ft 6 in. and the safety valves are located above the flood level, so these valves are not flooded.

In other areas except the main steam valve room, the fire suppression system is considered a flooding source. The flood water is drained to lower elevations through the drain system and openings. The potential flood level at this elevation is assumed as 0.15 m (6 in.).
The safety-related equipment and components are elevated above the flood level. Therefore, the Class 1E motor control center, switchgear, and remote shutdown panel are not flooded.

**Elevation 156 ft 0 in**

Flood water above El. 156 ft 0 in drains to the lower elevation through the floor drain and stairwells.

The equipment to be protected from flooding at El. 156 ft 0 in includes I&C equipment, cubicle coolers, and the console in main control room. The main control room area is protected from flooding in that no water lines are routed above or through the control room or computer room. Water lines routed to HVAC air handling units around the control room are contained in rooms with curbs that preclude the potential for water leakage from entering the control room or computer room. The MCR is also protected from the flooding sources outside the main control room with curbs and steel hole panel installed at the entrance of the MCR.

The following potential flooding sources are considered:

a. A postulated pipe failure of a moderate-energy line is considered.

b. A high-energy line break is not considered because there is no piping break in this area.

c. There is no break of non-seismic moderate-energy piping because piping in the auxiliary building is designed as seismic Category I or II.

d. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate is for firefighting equipment is assumed to be 0.044 m$^3$/s (700 gpm).

Based on flooding sources, the worst-case flooding scenario is rupture of a 0.10 m (4 in.) fire protection system line in a corridor. The flood water is drained to lower elevations through the drain system and openings. The potential flood level at this elevation is assumed as 0.61 m (2 ft) from the bottom El. 156 ft 0 in.

The safety-related equipment and components are elevated above El. 158 ft 0 in, so the I&C equipment, cubicle cooler, and consoles in the MCR are not flooded.
Elevation 174 ft 0 in

Flood water above El. 174 ft 0 in drains to the lower elevation through floor drains, stairwells, and openings. The EDG normal exhaust fan room is separated by distance and protected by flood barriers.

The equipment to be protected from flooding at El. 174 ft 0 in includes the control room supply air handling unit and EDG room normal supply air handling unit.

The following potential flooding sources are considered:

a. A postulated pipe failure of a moderate-energy line is considered.

b. A high-energy line break event is not considered because there is no piping break in this area.

c. There is no break of non-seismic moderate-energy piping because most of the piping in the auxiliary building is designed as seismic Category I or II.

d. Firefighting equipment represents internal flooding sources from at least the nearest two fire hose stations that could reach the fire zone. The discharge rate is for firefighting equipment is assumed to be 0.044 m$^3$/s (700 gpm).

Based on flooding sources, the worst-case flooding scenario is rupture of a 0.10 m (4 in.) fire protection system line in the general access area. The flood water is drained to lower elevations through the drain system and openings. The potential flood level at this elevation is assumed as 0.15 m (6 in.) from the bottom El. 174 ft 0 in.

The safety-related equipment and components are located above the flood level. Therefore, the control room supply AHUs, control room emergency makeup ACUs, and EDG room normal supply AHUs are not flooded.

3.4.1.5.3 Emergency Diesel Generator Building

The emergency diesel generator building is separated by distance from other buildings and divisionally separated by flood barriers. Emergency diesel generators (EDGs) are separated by distance and flood barriers so that an internal flooding event does not affect both EDGs simultaneously.
3.4.2 Analysis Procedures

Flood loads due to the design basis flood level and maximum groundwater level are estimated using the applicable codes and standards, as described in Section 3.8. Seismic Category I structures are designed to withstand flood loads and to remain watertight during the design basis flood event. The loads and load combinations provided in Section 3.8 take into consideration the static and dynamic loadings on seismic Category I structures including hydrostatic loading due to the design basis flood and/or the groundwater conditions specified in Table 2.0-1.

The COL applicant is to describe the basis for the Probable Maximum Flood (PMF) to determine the maximum site-specific ground water elevation above the grade that may occur from tsunami or hurricane sources (COL 3.4(9)).

The COL applicant is to identify any site-specific physical models that could be used to predict prototype performance of hydraulic structures and systems (COL 3.4(10)).

3.4.3 Combined License Information

COL 3.4(1) The COL applicant is to provide the site-specific design of plant grading and drainage.

COL 3.4(2) The COL applicant is to provide site-specific information on protection measures for the design basis flood, such levees, seawalls, flood walls, revetments or breakwaters or site bulkheads pursuant to NRC RG 1.102 as required in Subsection 2.4.10.

COL 3.4(3) The COL applicant is to establish procedures and programmatic controls to ensure the availability of the floor drainage.

COL 3.4(4) The COL applicant is to periodically inspect watertight doors and the penetration seals to ensure their functionality.

COL 3.4(5) The COL applicant is to provide flooding analysis with flood protection and mitigation features from internal flooding for the CCW Heat Exchanger Building and ESW Building.
COL 3.4(6) The COL applicant is to provide the site-specific flooding hazards from engineered features, such as water tank collapsing, water piping breaking, etc.

COL 3.4(7) The COL applicant is to confirm that the potential site-specific external flooding events are bounded by design basis flood values or otherwise demonstrate that the design is acceptable.

COL 3.4(8) The COL applicant is to provide the site-specific dewatering system if the plant is built below the design basis flood level.

COL 3.4(9) The COL applicant is to describe the basis for the Probable Maximum Flood (PMF) to determine the maximum site-specific ground water elevation above the grade that may occur from tsunami or hurricane sources.

COL 3.4(10) The COL applicant is to identify any site-specific physical models that could be used to predict prototype performance of hydraulic structures and systems.

3.4.4 References


### Reactor Containment Building Components Protected From Internal Flooding

<table>
<thead>
<tr>
<th>Item No.</th>
<th>Equipment No.</th>
<th>Equipment Description</th>
<th>Location</th>
<th>Building</th>
<th>Floor Elevation</th>
<th>SSC Level Relative to Flood Height</th>
<th>Flood Height above Floor m(ft)</th>
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<tbody>
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<td>ENFMS CH A DETECTOR ASSEMBLY</td>
<td>078C01</td>
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## Table 3.4-1 (3 of 58)

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| Item No. | Equipment No  | Equipment Description                        | Location | Building | Floor Elevation | SSC Level Relative to Flood Height | Flood Height above Floor  
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<td>Equipment Description</td>
<td>Location</td>
<td>Building</td>
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<td>Flood Height above Floor m(ft)</td>
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<th>Flood Height above Floor m(ft)</th>
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### Table 3.4-1 (7 of 58)

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## Table 3.4-1 (8 of 58)

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### APR1400 DCD TIER 2

Table 3.4-1 (12 of 58)

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## Table 3.4-1 (17 of 58)

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**APR1400 DCD TIER 2**

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**APR1400 DCD TIER 2**

Table 3.4-1 (23 of 58)

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**APR1400 DCD TIER 2**
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<td>596-V-2602</td>
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<td>REACTOR CNMT BLDG CHILLED WTR SUPPLY SIDE PENETRATION RELIEF VALVE</td>
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## APR1400 DCD TIER 2

### Table 3.4-1 (33 of 58)

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<th>Equipment Description</th>
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<th>Flood Height above Floor (m(ft))</th>
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## Table 3.4-1 (34 of 58)

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<th>Equipment Description</th>
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### Table 3.4-1 (36 of 58)

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Table 3.4-1 (40 of 58)

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# APR1400 DCD TIER 2

## Table 3.4-1 (46 of 58)

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<th>SSC Level Relative to Flood Height</th>
<th>Flood Height above Floor</th>
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# APR1400 DCD TIER 2

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### Table 3.4-1 (55 of 58)

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### Table 3.4-1 (56 of 58)

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<th>Floor Elevation</th>
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### Table 3.4-1 (57 of 58)

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<th>Equipment Description</th>
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<th>Building</th>
<th>Floor Elevation</th>
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<td>441-J-PT-0311D</td>
<td>SIT TANK 4 TK-01D PRESSURE (WIDE)</td>
<td>156C01</td>
<td>RCB</td>
<td>156'-0&quot; above 0.15(0.5)</td>
<td>156'-0&quot; above 0.15(0.5)</td>
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<td>441-J-PT-0321B</td>
<td>SIT TANK 2 TK-01B PRESSURE (WIDE)</td>
<td>156C01</td>
<td>RCB</td>
<td>156'-0&quot; above 0.15(0.5)</td>
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<td>441-J-PT-0331C</td>
<td>SIT TANK 3 TK-01C PRESSURE (WIDE)</td>
<td>156C01</td>
<td>RCB</td>
<td>156'-0&quot; above 0.15(0.5)</td>
<td>156'-0&quot; above 0.15(0.5)</td>
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<td>1086</td>
<td>441-J-PT-0341A</td>
<td>SIT TANK 1 TK-01A PRESSURE (WIDE)</td>
<td>156C01</td>
<td>RCB</td>
<td>156'-0&quot; above 0.15(0.5)</td>
<td>156'-0&quot; above 0.15(0.5)</td>
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<td>1087</td>
<td>521-J-PT-1013A</td>
<td>S/G 1 PRESSURE CH A</td>
<td>156C01</td>
<td>RCB</td>
<td>156'-0&quot; above 0.15(0.5)</td>
<td>156'-0&quot; above 0.15(0.5)</td>
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<td>1088</td>
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<td>S/G 1 PRESSURE CH B</td>
<td>156C01</td>
<td>RCB</td>
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<td>156'-0&quot; above 0.15(0.5)</td>
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<td>S/G 1 PRESSURE CH C</td>
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<td>156'-0&quot; above 0.15(0.5)</td>
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### Table 3.4-1 (58 of 58)

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<th>Equipment No</th>
<th>Equipment Description</th>
<th>Location</th>
<th>Building</th>
<th>Floor Elevation</th>
<th>SSC Level Relative to Flood Height</th>
<th>Flood Height above Floor m(ft)</th>
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<td>521-J-PT-1023A</td>
<td>S/G 2 PRESSURE CH A</td>
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<td>RCB</td>
<td>156'-0&quot; above</td>
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<td>521-J-PT-1023B</td>
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<td>521-J-PT-1023C</td>
<td>S/G 2 PRESSURE CH C</td>
<td>156C01</td>
<td>RCB</td>
<td>156'-0&quot; above</td>
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<td>521-J-PT-1023D</td>
<td>S/G 2 PRESSURE CH D</td>
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<td>CNMT UPPER OPERATING AREA MONITOR</td>
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<td>RCB</td>
<td>156'-0&quot; above</td>
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<td>CNMT UPPER OPERATING AREA MONITOR</td>
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<td>763-J-TE-0031A</td>
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<td>RCB</td>
<td>156'-0&quot; above</td>
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<td>RCB</td>
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(1) These components are not required to be protected against internal flooding because they are not failed due to submergence or the loss of their functions does not impact on safe shutdown.
### Table 3.4-2 (1 of 281)

#### Auxiliary Building Components Protected from Internal Flooding

<table>
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<tr>
<th>Item No.</th>
<th>Equipment No.</th>
<th>Equipment Description</th>
<th>Location</th>
<th>Building</th>
<th>Floor Elevation</th>
<th>Flood Elevation</th>
<th>Flood Drain [inch(EA)]</th>
<th>Emergency Overflow Line [inch(EA)]</th>
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<td>1</td>
<td>442-M-HE02A</td>
<td>CS MINIFLOW HX 1</td>
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<td>Auxiliary</td>
<td>50'-0&quot;</td>
<td>(1) 64'-0&quot;</td>
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<td>6&quot; EOL (1)</td>
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<td>2</td>
<td>442-M-OR01A</td>
<td>RESTRICTION ORIFICE</td>
<td>050A01C</td>
<td>Auxiliary</td>
<td>50'-0&quot;</td>
<td>(1) 64'-0&quot;</td>
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<td>3</td>
<td>442-M-PP01A</td>
<td>CONTAINMENT SPRAY PUMP 1</td>
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<td>(1) 64'-0&quot;</td>
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<td>4</td>
<td>606-M-HV10A</td>
<td>CS PUMP &amp; MINIFLOW HEAT EXCHANGER RM</td>
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<td>(1) 64'-0&quot;</td>
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<td>11</td>
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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

## Table 3.4-2 (41 of 281)

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

### Table 3.4-2 (46 of 281)

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## Table 3.4-2 (54 of 281)

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### APR1400 DCD TIER 2

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**Flood Height**

- **Relative to**: Above reference surface
- **Above**: Above ground level
### APR1400 DCD TIER 2

#### Table 3.4-2 (57 of 281)

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<th>Item No.</th>
<th>Equipment No.</th>
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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

### Table 3.4-2 (59 of 281)

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

Table 3.4-2 (65 of 281)

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

### Table 3.4-2 (69 of 281)

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

## Table 3.4-2 (79 of 281)

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# APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

## Table 3.4-2 (85 of 281)

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3.4-167
### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

### Table 3.4-2 (110 of 281)

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### APR1400 DCD TIER 2

#### Table 3.4-2 (111 of 281)

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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Table 3.4-2 (129 of 281)

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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Table 3.4-2 (135 of 281)
## APR1400 DCD TIER 2

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<th>Flood Elevation</th>
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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### Table 3.4-2 (142 of 281)

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**APR1400 DCD TIER 2**
# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

## Table 3.4-2 (146 of 281)

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### APR1400 DCD TIER 2

#### Table 3.4-2 (148 of 281)

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<th>Flood Elevation</th>
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# APR1400 DCD TIER 2

## Table 3.4-2 (150 of 281)

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### APR1400 DCD TIER 2

#### Table 3.4-2 (152 of 281)

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| APR1400 DCD TIER 2 | Table 3.4-2 (152 of 281) |

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### APR1400 DCD TIER 2
### APR1400 DCD TIER 2

#### Table 3.4-2 (156 of 281)

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<th>Equipment No.</th>
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<td>SSC Level Relative to Flood Height</td>
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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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<th>Flood Elevation</th>
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**APR1400 DCD TIER 2**

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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3.4-261

Rev. 3
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## Table 3.4-2 (187 of 281)

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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**3.4-282**  
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## APR1400 DCD TIER 2

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**APR1400 DCD TIER 2**

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## APR1400 DCD TIER 2

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## APR1400 DCD Tier 2

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## APR1400 DCD TIER 2

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## Table 3.4-2 (219 of 281)

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<th>Item No.</th>
<th>Equipment No.</th>
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<th>Flood Elev.</th>
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## APR1400 DCD TIER 2

Table 3.4-2 (220 of 281)

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<th>Flood Elevation</th>
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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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Table 3.4-2 (225 of 281)
### APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

#### Table 3.4-2 (234 of 281)

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<th>Item No.</th>
<th>Equipment No.</th>
<th>Equipment Description</th>
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<th>Building</th>
<th>Floor Elevation</th>
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<th>Flood Elevation</th>
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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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<td>Flood Elevation</td>
<td>Flood Drain [inch(EA)]</td>
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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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### Table 3.4-2 (262 of 281)

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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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#### Table 3.4-2 (268 of 281)

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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# APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

### Table 3.4-2 (277 of 281)

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## APR1400 DCD TIER 2

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(1) These components are protected by flood barrier against internal flooding. And these components are not required to be protected from flooding sources due to redundancy of other trains/components.
Figure 3.4-1  Location of Watertight Doors and Flood Barrier Plan View Elevation 55'-0"
Figure 3.4-2  Location of Watertight Doors and Flood Barrier Plan View Elevation 78'-0"
Figure 3.4-3  Location of Watertight Doors and Flood Barrier Plan View Elevation 100'-0"
Figure 3.4-4  Location of Watertight Doors and Flood Barrier Plan View Elevation 120'-0"
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Figure 3.4-5 Location of Watertight Doors and Flood Barrier Plan View Elevation 137'-6"
Figure 3.4-6  Location of Watertight Doors and Flood Barrier Plan View Elevation 156'-0"
Figure 3.4-7  Location of Watertight Doors and Flood Barrier Plan View Elevation 174'-0"
3.5 Missile Protection

In accordance with 10 CFR Part 50, Appendix A, GDC 2 and 4 (Reference 1), essential structures, systems, and components (SSCs) important to safety are required to be protected from internal and external missiles.

Missile protection is provided for SSCs important to safety so that internal and external missiles do not cause the release of significant amounts of radioactivity or prevent the safe shutdown of the reactor considering a single failure.

The protection of SSCs important to safety is accomplished by one or more of the following:

a. Minimizing the sources of missiles by equipment design features that prevent missile generation

b. Orienting or physically separating potential missile sources away from safety-related equipment and components

c. Containing the potential missiles through the use of protective shields or barriers near the missile source or safety-related facility and equipment

d. Hardening of safety-related equipment and components to withstand missile impact when such impacts cannot be reasonably avoided by the methods listed above

Table 3.2-1 is the list of SSCs. Essential SSCs outside reactor containment building to be protected from externally generated missiles are provided in Table 3.5-4. SSCs located inside the seismic Category I containment building are protected from missiles outside reactor containment building by thick concrete walls and are therefore omitted. General arrangement drawings showing locations of the SSCs are given in Section 1.2.

3.5.1 Missile Selection and Description

For equipment with energy sources capable of generating a missile, the selection is based on the application of a single failure criterion to the retention features of the component. Where sufficient redundancy of retention features is provided to a component, then no missile impact is postulated due to a single failure of a retention feature of that component.
The types of missiles considered in the design of safety-related SSCs are categorized as follows, based on their origin:

a. Internally generated missiles (outside the containment)
b. Internally generated missiles (inside the containment)
c. Turbine missiles
d. Natural phenomena missiles
e. Site proximity missiles (except aircraft)
f. Aircraft hazards

The criteria for each type of missiles are described below. An evaluation is performed to verify that missiles do not adversely affect the safety functions of safety-related SSCs.

3.5.1.1 Internally Generated Missiles (Outside the Containment)

The criteria used for protection from internally generated missiles outside the containment are generally consistent with the NRC guidelines in Standard Review Plan (SRP) 3.5.1.1 (Reference 2).

Barriers or retention features in structures other than containment are designed in accordance with the design procedure described in Subsection 3.5.3. The following procedures are used to achieve conformance with design criteria for protection against missiles that are generated by the failure of pressurized or rotating components outside the containment.

a. Classifying missiles based on their source

b. Classifying equipment and components that require protection from missiles, based on the design criteria described in Section 3.5

c. Following the procedures listed below to protect equipment and components from failure

1) Identifying features of missiles generated by the failure of rotating components (e.g., motor-driven pumps, fans), pressurized components (e.g.,
reactor vessel closure head stud and nut), or gravitational missiles (e.g., falling objects resulting from a nonseismically designed SSC during a seismic event)

2) Determining the effects of the identified missiles on equipment and components (design structures between the missiles and protected equipment to withstand the missile impact if the structures are in the trajectory path)

3) Relocating equipment and missile generation source in relation to postulated missile trajectories to protect equipment from being impacted by a missile

4) Installing barriers or retention features, to protect equipment and components from impact by a missile.

Internally generated missiles can be generated from two types of equipment: rotating components and pressurized components. Rotating components include turbine wheels, fans, and pumps. Pressurized components include valve stems, valve bonnets, bolted connections on pressure vessels, and instrument wells.

Missiles generated by postulated failures of rotating components are selected and evaluated based on the following conditions:

a. All rotating components that are operated during normal operating plant conditions are capable of generating missiles.

b. The energy of potential missiles generated from rotating components is based on a 120 percent overspeed condition as a minimum.

c. The missile is postulated to occur only if the energy of the missile is sufficient to perforate the equipment’s protective housing.

Missiles generated by postulated failures of pressurized components are selected and evaluated based on the following conditions:

a. Pressurized components in the systems whose maximum operating pressure exceeds 19.3 kg/cm² (275 psig) are assumed to be missile generation sources.
b. Connecting portions installed on piping or components are assumed to be missile
generation sources. Connecting portions include thermowells, pressure gauges,
and lines for vents, drains, and testing.

c. A connecting portion may be eliminated as a missile generation source if it is
welded and its design strength is stronger than base metal.

d. Valves with bolted bonnets are most commonly used valve type such as gate,
check or globe valves in high energy piping. The body and bolted bonnet of
these valves constructed with ASME Section III or ASME B16.34 are unlikely to
become missile sources due to the limitation of stresses in the bonnet-to-body
bolting material by rules set forth in ASME Code. Even if a bonnet-to-body bolt
failure were to occur, the likelihood of all bolts experiencing a simultaneous
complete failure is very remote. The widespread use of valves with bolted
bonnet in nuclear industry and the low historical incidence of complete valve
bonnet failures demonstrate that the valves with bolted bonnets type need not be
considered as credible missile sources.

e. Pressure seal bonnet type valves are also constructed in accordance with ASME
Section III or ASME B16.34. The valve bonnets are prevented from becoming
missiles by the retaining ring. If retaining ring were to fail in shear, then the
yoke would capture the bonnet or at least significantly reduce its energy.
Because of the combination of these design features bonnet ejection incident is
highly improbable, and hence bonnets are not considered missile sources.

f. The design feature of threaded valve stem with face hardened backseats prevents
the ejection of stems. The stems are prevented from becoming credible missiles.
And the stems having valve actuators are additionally restrained by the valve
actuators.

g. Nuts, bolts, nut-and-bolt combinations, and nut-and-stud combinations need not be
considered as credible missile sources because it has not enough energy to eject a
missile.

h. Non-ASME pressurized vessels with an operating pressure greater than
19.3 kg/cm² (275 psig) are considered missile generation sources. ASME vessels
are not considered missile generation sources because they are designed,
fabricated, examined, and tested in accordance with ASME Section III.
i. Non-ASME valves in piping systems with an operating pressure greater than 19.3 kg/cm² (275 psig) are considered missile generation sources.

j. An industrial pressure bottle containing highly pressurized gas is considered a missile generation source except when the bottle is designed with overpressure protection and is located in a separate room to control the effect of an explosion.

Internally generated missiles (outside the containment) from rotating and pressurized components are not considered credible in accordance with the criteria described above.

Missiles falling from heavy load transfers by crane and missiles from dropped SSCs designed to non-seismic category outside the containment are considered gravity-based missiles. The design conforms with SRP 9.1.5 (Reference 3) and NUREG-0612 (Reference 4) for falling heavy loads from equipment or component transfers, and the design provides reasonable assurance that the effect of heavy load drops during transfers by a crane is eliminated by blocking the path above the systems and components that are necessary to achieve a safe shutdown or accident. The drop of nonseismically designed SSCs outside the containment could affect safety-related systems. Therefore, they are designed to seismic Category II to protect the safety-related systems from the impact of dropped objects.

The COL applicant is to provide the procedure for heavy load transfer to strictly limit the transfer route inside and outside containment during plant maintenance and repair periods (COL 3.5(1)).

3.5.1.1.1 Potential Missiles from Rotating Component (Outside Containment)

If the probability of missile generation $P_1$ is maintained less than $10^{-7}$ per year, missile is not considered statistically significant. If the probability of occurrence is greater than $10^{-7}$ per year, the probability of impact on a significant target is determined. If the product of these two probabilities is less than $10^{-7}$ per year, the missile is not considered statistically significant.

All rotating components such as pumps and motors are considered rotating missile generation sources outside containment. However, there is no postulated missile because $P_1$ is less than $10^{-7}$ per year for the following reasons.
a. Pump motors are an induction type that have relatively slow running speeds and
are not prone to overspeed. These motors are pretested at full running speed by
the motor vendor prior to installation.

b. The motor stator serves as a natural container of rotor missiles if any are generated.

c. Safety-related pumps have relatively low suction pressures and are not driven to
overspeed due to a pipe break in their discharge lines. In addition, the induction
motor would act as a brake to prevent pump overspeed.

d. Industry pumps are designed to prevent the penetration of pump casings from
impeller pieces under overspeed conditions through vendor demonstration that the
supplied pump casing is adequate to retain postulated fragments.

The turbine building, which contacts with the seismic Category I auxiliary building, is
designed as seismic Category II. The turbine building does not contain safety-related
systems or components and does not require design for protection from rotating and
pressurized components that become missiles.

The turbine, the object with the largest kinetic energy in the turbine building, is considered
a missile generation source. Turbine missiles are described in Subsection 3.5.1.3. The
main feedwater pump, the object with the second largest kinetic energy outside containment,
is considered to be a generation source for a missile that flies toward the auxiliary building,
even though the main feedwater pump is designed such that the inside fragment cannot
perforate the casing. By assuming penetration of its casing by a fragment, the results
provide reasonable assurance that missiles from the main feedwater pump would not
perforate the external wall of the auxiliary building. Considering that missiles that are
generated from rotating components near the auxiliary building have rotors that are oriented
toward the auxiliary building, reasonable assurance of the protection of safety-related
systems and components inside the auxiliary building is provided.

3.5.1.1.2 Potential Missiles from Pressurized Components (Outside Containment)

If the probability of missile generation $P_1$ is maintained less than $10^{-7}$ per year, the missile
is not considered statistically significant. If the probability of occurrence is greater than
$10^{-7}$ per year, the probability of impact on a significant target is determined. If the product
of these two probabilities is less than $10^{-7}$ per year, the missile is not considered statistically
significant.
Valves are considered potential missiles from pressurized components outside containment. However, no postulated missiles are generated by valves because $P_1$ is less than $10^{-7}$ per year for one or more of the following reasons:

a. All valve stems are provided with a backseat or shoulder larger than the valve bonnet opening.

b. Motor-operated and manual valve stems are restrained by stem threads.

c. Operators on motor, hydraulic, and pneumatic operated valves prevent stem ejection.

d. Pneumatic-operated diaphragms and safety valve stems are restrained by the actuator casing.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The criteria used for protection of internally generated missiles inside the containment are generally consistent with the guidelines in SRP 3.5.1.2 (Reference 5).

Structures inside the containment, including the secondary shield wall, refueling pool wall, structural beams, and floor slabs, serve as missile shields for equipment, including the reactor coolant loop, that must be protected from missiles. These structures protect the reactor coolant pressure boundary (RCPB), containment liner plate, containment penetration, main steam line, main feedwater line, direct vessel injection line, and steam generator, including the instrument connection of steam side, blowdown line, and drain line from the missile. Engineered safety features, except for some portions of piping for direct vessel injection following a LOCA, are located outside the secondary shield wall to minimize the effects from missiles generated by the RCPB.

The protective shield is installed above the control element drive mechanism to protect the control rod drive mechanism and reactor vessel and SSCs required for safe shutdown from internally generated missiles and a missile from the control element drive mechanism breakaway.

The protective features (such as structural concrete walls, structural steel beams, or floor slabs) inside the containment are provided to protect the containment liner plate, isolation system, and the main steam system related to the steam generator from missiles caused by
the main steam and main feedwater line breaks, and to prevent the malfunction of other systems or equipment.

The secondary shield wall between the containment wall and refueling pool wall serves as a shield to protect the reactor coolant system, including the steam generator, from missiles generated inside the containment annulus area. The secondary shield wall also serves as a shield to protect safe-shutdown equipment, such as the safety injection tank, from missiles generated by the RCPB.

Missiles falling from heavy load transfers by crane and missiles from dropped SSCs designed to non-seismic category inside the containment are considered gravity-caused missiles. Designs for other lifts comply only with SRP 9.1.5 and NUREG-0612 for falling heavy loads from equipment or component transfers, and the design provides reasonable assurance that the effect of heavy load drops during transfers by crane is eliminated by blocking the path above the systems and components that are necessary to achieve a safe shutdown or protect against an accident. The drop of nonseismically designed SSCs in the containment could affect safety-related systems. Therefore, they are designed to seismic Category II to protect the safety-related systems from the impact of dropped objects.

The COL applicant is to provide the procedure for heavy load transfer to strictly limit the transfer route inside and outside containment during plant maintenance and repair period (COL 3.5(1)).

The COL applicant is to provide the procedures which ensure that equipment required during maintenance, should be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSC important to safety, or seismically restrained (COL 3.5(2)).

3.5.1.2.1 Potential Missiles from Rotating Components (Inside Containment)

If the probability of missile generation $P_1$ is maintained less than $10^{-7}$ per year, the missile is not considered statistically significant. If the probability of occurrence is greater than $10^{-7}$ per year, the probability of impact on a significant target is determined. If the product of these two probabilities is less than $10^{-7}$ per year, the missile is not considered statistically significant.

Rotating components inside the containment building are reactor coolant pumps, heating, ventilation and air conditioning (HVAC) equipment, and pump impellers. However, they
are not postulated missile sources, because \( P_1 \) is less than \( 10^{-7} \) per year for the reasons described in Subsection 3.5.1.1.1.

3.5.1.2.2 Potential Missiles from Pressurized Components (Inside Containment)

Table 3.5-1 lists the postulated missiles and their weight, shape, dimensions, and impact energy. Major pretensioned studs and nuts, instruments, and the control rod drive mechanism missiles are included. Items that are excluded because of redundant retention features are valve stems, valve bonnets, and pressurized cover plates.

Credible missiles resulting from failures of pressurized components are selected based on the same conditions as listed in Subsection 3.5.1.1.

Potential missiles might be generated from pressurized components installed in the high-energy piping system. Most of the valves installed on a high-energy line designed, constructed, and tested in accordance with ASME B16.34 with Class 900 are excluded as a missile source from pressurized components. Valves installed on auxiliary steam systems are also excluded as a missile source because the operating pressure is below 19.3 kg/cm\(^2\) (275 psig).

3.5.1.3 Turbine Missiles

Although the auxiliary system associated with the turbine is non-safety-related, missiles generated by turbine failure can adversely affect the integrity of essential SSCs as defined in Regulatory Guide 1.115 (Reference 13) Appendix A. Table 3.5-4 lists the essential SSCs outside the reactor building that are evaluated to provide reasonable assurance that they are adequately protected from potential turbine missiles. None of the essential SSCs listed are within the low-trajectory missile strike zone.

3.5.1.3.1 Geometry

The turbine generator is composed of one high-pressure and three low-pressure turbines. As shown in Figure 3.5-1, the turbine shaft is placed in a line with the containment and auxiliary building. The figure shows that the turbine generator is placed with favorable orientation so that all essential the SSCs are excluded from the low-trajectory turbine missile strike zone, as defined by Regulatory Guide 1.115, and are concentrated in an area bounded by lines inclined at 25 degrees to the turbine wheel planes and passing through the end wheels of the low-pressure stages. The arrangement is selected to meet Regulatory
Guide 1.115 approach C.2.a. An assessment of the orientation of the turbine generator of this and other unit(s) at multi-unit sites for the probability of missile generation using the evaluation in Subsection 3.5.1.3.2 shall verify essential SSCs are outside the low-trajectory turbine missile strike zone.

3.5.1.3.2 Evaluation

Essential SSCs are designed to be protected from potential turbine missiles. The design criteria for the protection of essential SSCs are described in Section 3.5.

The probability of unacceptable damage resulting from failure of a single turbine (P4) is calculated by multiplying all items (P1 × P2 × P3) below.

a. Probability of turbine failure resulting in the ejection of turbine disc (or internal structure) fragments through the turbine casing (P1)

b. Probability of ejected missiles perforating intervening barriers and striking essential SSCs (P2)

c. Probability of essential SSCs that are struck failing to perform their safety functions (P3)

Thus, the total damage probability per year (P) of each unit resulting from turbine failure in site is calculated as follows.

$$P = P_4 = P_1 \times P_2 \times P_3$$

Where P1, P2, and P3 are as defined above, P3 is assumed to be 1.0 (i.e., if struck, an SSC is assumed to fail), and P4 is the probability of unacceptable damage to safety-related SSCs in target unit.

As described in SRP 3.5.1.3 (Reference 6), on the basis of simple estimates for a variety of plant layouts, the strike and damage probability product (P2 x P3) can be reasonably assumed to fall within a range that depends on the gross features of turbine generator orientation.

For favorably oriented turbine generators, the product of P2 and P3 tends to be in the range of $10^{-4}$ to $10^{-3}$ per year per plant.
An estimate of $10^{-3}$ per year for a favorable orientation is conservatively assumed.

Favorable turbine generator placement and orientation, combined with the design and fabrication processes, the redundant and fail-safe turbine control system, maintenance and inspection programs as described in Section 10.2, and overspeed protection systems, provide an acceptably small probability of turbine missiles causing damage to essential SSCs, $P_4$.

The probability of turbine missile generation $P_1$ shall be verified by ITAAC to be less than $1 \times 10^{-5}$ per year for a favorable orientation to demonstrate that the probability of failure of essential SSCs from turbine missile $P_4$ is less than $1 \times 10^{-7}$ per year.

In addition, ITAAC shall provide verification that the as-built turbine material properties, turbine rotor and blade designs, pre-service inspection and testing results and in-service testing and inspection requirements meet the requirements defined in the turbine missile probability analysis.

The COL applicant is to perform an assessment of the orientation of the turbine generator of this and other unit(s) at multi-unit sites for the probability of missile generation using the evaluation of Subsection 3.5.1.3.2 to verify that essential SSCs are outside the low-trajectory turbine missile strike zone (COL 3.5(3)).

3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

Safety-related SSCs of the APR1400 are protected against the impact generated by tornado or hurricane missiles. The protection measures consist of seismic Category I structures, shields, and barriers to withstand the effects of missile impact generated by a tornado or hurricane. The protection provides reasonable assurance of conformance with 10 CFR Part 50, Appendix A, GDC 2 and 4 and 10 CFR 52.47(b)(1) (Reference 7).

Procedures are followed to predict local damage of the impacted area and the response of seismic Category I structures, shields and barriers, or portions thereof, to the missile impact generated by a tornado or hurricane. The procedure for determining local damage includes an estimation of the depth of penetration and, in the case of concrete barriers, the potential for generation of secondary missiles by spalling or scabbing.
Procedures for overall response include assumptions on acceptable ductility ratios where elasto-plastic behavior is relied upon and methods for estimation of forces, moments, and shears induced in the barrier by the impact force of the missile.

Design basis missile spectra for tornadoes and their maximum horizontal speeds (vertical speeds are 67 percent of horizontal speeds) are determined based on the Region I spectrum, which is defined in Table 2 of NRC RG 1.76 (Reference 8). Design basis missile spectra for hurricanes are determined based on Table 1 of NRC RG 1.221 (Reference 9). The design basis missile velocities for hurricanes correspond to the hurricane windspeed of 116 m/s (260 mph) in Table 2 of NRC RG 1.221. The selected missile types are listed below, and the specified dimensions, mass, and velocity of each missile are defined in Table 3.5-2.

- a. Pipe – Rigid missile that tests penetration
- b. Automobile – Massive high-kinetic-energy missile that deforms on impact
- c. Solid steel sphere – Small rigid missile of sufficient size to pass through any opening in protective barriers

Exterior walls and roof slabs of seismic Category I structures are required to function as missile barriers. The design of missile barriers provides reasonable assurance that the structure will not collapse under the missile load and the barrier will not be penetrated.

The COL applicant is to evaluate site-specific hazards induced by external events that may produce more energetic missiles than tornado or hurricane missiles, and provide reasonable assurance that the seismic Category I structures are designed to withstand these loads (COL 3.5(4)).

The COL applicant is to confirm that automobile missiles cannot be generated within a 0.5 mile radius of safety-related SSCs that would lead to impact higher than 10.06 m (33 ft) above plant grade (COL 3.5(5)).

The COL applicant is to identify applicable tornado missile spectra and associate velocities for the compound building, and to evaluate the missile protection provided by the building (COL 3.5(6)).
3.5.1.5 Site Proximity Missiles (Except Aircraft)

The COL applicant is to evaluate the potential for site proximity explosions and missiles due to train explosions (including rocket effects), truck explosions, ship or barge explosions, industrial facilities, pipeline explosions, or military facilities (COL 3.5(7)). If the total probability of explosion is greater than an order of magnitude of $10^{-7}$ per year, a missile description, including size, shape, weight, energy, material properties, and trajectory, will be specified. A description of the missile effects on the SSCs will be developed and addressed, if necessary.

3.5.1.6 Aircraft Hazards

The COL applicant is to provide justification for the site-specific aircraft hazard and an aircraft hazard analysis in accordance with the requirements of NRC RG 1.206 (Reference 10) (COL 3.5(8)).

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

All safety-related SSCs required to safely shut the reactor down and maintain it in a safe condition are housed in seismic Category I structures. Seismic Category I structures are designed as tornado/hurricane resistant (see Subsection 3.5.1.4) and other external missile resistant.

Structures used to protect safety-related SSCs meet the requirements of NRC RGs 1.13 (Reference 11), 1.27 (Reference 12), 1.115 (Reference 13), and 1.117 (Reference 14).

Essential SSCs protected against missile impact are listed in Table 3.5-4. SSCs inside containment are protected by the thick seismic Category I concrete walls of the reactor building and are not listed.

Openings and penetrations through the exterior walls and roofs of seismic Category I structures and the location of equipment in the vicinity of such openings are arranged so that a missile passing through the opening would not prevent the safe shutdown of the plant and would not result in an offsite release of nuclides exceeding 25% of the limits defined in 10 CFR Part 100 (Reference 15). Otherwise, structural barriers composed of enclosures, missiles-resistant doors and covers, and physical protection features are designed to resist tornado and hurricane missiles in accordance with the design procedures described in
Subsection 3.5.3. Tornado and hurricane missiles are not postulated to strike more than once at a target location. Because of the robustness of the exterior wall, all seismic Category I structures are capable of withstanding the impact of each identified missile.

The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for missile loads will not prevent safety-related SSCs from performing their safety function (COL 3.5(9)).

3.5.3 Barrier Design Procedures

Missile barriers, whether steel or concrete, are designed with sufficient strength and thickness to prevent local damage including perforation, spalling and scabbing, and overall damage. The procedures by which structures and barriers are designed to perform this function are presented in this subsection.

There are no composite sections designed as missile barrier in APR1400. Containment liner plate is not designed as missile barriers nor composite sections.

3.5.3.1 Evaluation of Local Structural Effects

The prediction of local damage in the immediate vicinity of an affected area depends on the basic material of construction of the barrier itself. Corresponding procedures are described below.

3.5.3.1.1 Concrete Barriers

Local damage prediction for concrete structures includes the estimation of the depth of missile penetration and an assessment of whether secondary missiles could be generated by spalling. Design criteria for concrete barriers are consistent with the National Defense Research Council (NDRC), “A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects” (Reference 16). The modified NDRC formula is used to estimate the missile penetration depth, and barrier thickness to prevent perforation, spalling, and scabbing effects. The design thicknesses of missile barriers are 20 percent greater than the threshold values for the phenomenon being prevented. The design thicknesses also satisfy the minimum acceptable barrier thickness requirements for local damage prediction against tornado-generated missiles as well as hurricane-generated missiles. The minimum barrier thicknesses for local damage due to
tornado- and hurricane-generated missiles are provided in Table 3.5-3. The equations used to evaluate local structural effects are described as follows.

3.5.3.1.1.1 Penetration

The depth of missile penetration, \( x \), is calculated using the modified NDRC formulas:

\[
x = \begin{cases} 
K N W_m \left( \frac{v_o}{1000d} \right)^{1.80} + d, & \text{if } \frac{x}{d} > 2.0, \\
\sqrt{4KNW_md \left( \frac{v_o}{1000d} \right)^{1.80}}, & \text{if } \frac{x}{d} > 2.0
\end{cases}
\]

Where:

- \( x \) = penetration depth (inch)
- \( K \) = concrete penetrability factor, based on experimental data
  \[ K = \frac{180}{\sqrt{f_c'}}, \text{ (with } f_c' \text{ in psi)} \]
- \( N \) = missile shape factor and is taken as follows:
  - 0.72 for flat-nosed bodies
  - 0.84 for blunt-nosed bodies
  - 1.00 for spherical end
  - 1.14 for very sharp nose
- \( W_m \) = missile weight (lb)
- \( V_o \) = missile impact velocity (ft/sec)
- \( d \) = effective missile diameter; for non-solid cylindrical missiles or solid missiles with non-circular cross section, \( d \) is the diameter of an equivalent solid cylindrical shaped missile with the same contact surface area as the actual missile (inch)
3.5.3.1.1.2 Perforation

The thickness, $t_p$, required to prevent perforation is calculated by the modified NDRC formulas:

$$t_p = \left[ 3.19 \left( \frac{X}{d_e} \right) - 0.178 \left( \frac{X}{d_e} \right)^2 \right] d_e, \quad \text{for } \frac{X}{d_e} \leq 1.35$$

$$t_p = \left[ 1.32 + 1.24 \left( \frac{X}{d_e} \right) \right] d_e, \quad \text{for } 1.35 \leq \frac{X}{d_e} \leq 13.5$$

Where:

- $t_p$ = barrier thickness to prevent perforation (inch)
- $X$ = penetration depth (inch)
- $d_e$ = diameter of a solid bar with the same contact area as the pipe, effective missile diameter for solid missile (inch)

3.5.3.1.1.3 Scabbing

The thickness $t_s$, required to preclude scabbing is calculated by the modified NDRC formulas:

$$\frac{t_s}{d_e} = \left[ 7.91 \left( \frac{X}{d_e} \right) - 5.06 \left( \frac{X}{d_e} \right)^2 \right], \quad \text{for } \frac{X}{d_e} \leq 0.65$$

$$\frac{t_s}{d_e} = 2.12 + 1.36 \left( \frac{X}{d_e} \right), \quad \text{for } 0.65 \leq \frac{X}{d_e} \leq 11.75$$

Where:

- $t_s$ = barrier thickness to prevent scabbing (inch)
- $X$ = penetration depth (inch)
- $d_e$ = diameter of a solid bar with the same contact area as the pipe, effective missile diameter for solid missile (inch)
3.5.3.1.2 Steel Barriers

Steel barriers are not used in the APR1400 design to protect the safety-related equipment or systems from missile impact.

3.5.3.2 Overall Damage Prediction

Overall evaluation of structures/barriers under impact or impulsive loads is focused on providing reasonable assurance that the structures or barriers will not collapse and have excessive deformations to affect the function of safe shutdown equipment. For the evaluation of overall response of reinforced concrete barriers under impact and impulse loads, nonlinear and elasto-plastic response of structures is used.

Evaluations of the overall damage due to missile impact are performed by either considering missile impact in the elastic range of the structural element with other loadings applied and accounting for rebound effects of the impact, or by assuming that the inelastic capacity of the structural element resists missile impact loads. Inelastic impact analyses are performed by assuming that the full elastic capacity of the structural element is used to accommodate other loading conditions, and that the missile impact loads are accommodated in-elastically based on the ductility of the structural element. Code requirements for ductility shall be satisfied for the structural response evaluations under the missile impact.

Excessive deformation is limited by the allowable ductility ratios, which is defined as the ratio of maximum displacement to the yield displacement of structural element, where the maximum displacement will not result in the loss and damage of structural elements and components. The ductility limits for concrete structures or barriers for various loading categories are defined in Table 3.5-5 which is based on NRC RG 1.142 (Reference 18).

3.5.4 Combined License Information

COL 3.5(1) The COL applicant is to provide the procedure for heavy load transfer to strictly limit the transfer route inside and outside containment during plant maintenance and repair periods.

COL 3.5(2) The COL applicant is to provide the procedures which ensure that equipment required during maintenance, should be removed from...
containment prior to operation, moved to a location where it is not a potential hazard to SSC important to safety, or seismically restrained.

COL 3.5(3) The COL applicant is to perform an assessment of the orientation of the turbine generator of this and other unit(s) at multi-unit sites for the probability of missile generation using the evaluation of Subsection 3.5.1.3.2 to verify that essential SSCs are outside the low-trajectory turbine missile strike zone.

COL 3.5(4) The COL applicant is to evaluate site-specific hazards induced by external events that may produce more energetic missiles than tornado or hurricane missiles, and provide reasonable assurance that the seismic Category I structures are designed to withstand these loads.

COL 3.5(5) The COL applicant is to confirm that automobile missiles cannot be generated within a 0.5 mile radius of safety-related SSCs that would lead to impact higher than 10.06 m (33 ft) above plant grade.

COL 3.5(6) The COL applicant is to identify applicable tornado missile spectra and associate velocities for the compound building, and to evaluate the missile protection provided by the building.

COL 3.5(7) The COL applicant is to evaluate the potential for site proximity explosions and missiles due to train explosions (including rocket effects), truck explosions, ship or barge explosions, industrial facilities, pipeline explosions, or military facilities.

COL 3.5(8) The COL applicant is to provide justification for the site-specific aircraft hazard and an aircraft hazard analysis in accordance with the requirements of NRC RG 1.206.

COL 3.5(9) The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for missile loads will not prevent safety-related SSCs from performing their safety function.

3.5.5 References


15. 10 CFR Part 100 “Reactor Site Criteria,” U.S. Nuclear Regulatory Commission.


Table 3.5-1 (1 of 2)

Kinetic Energy of Potential Missiles

Security-Related Information – Withhold Under 10 CFR 2.390
Security-Related Information – Withhold Under 10 CFR 2.390
### Table 3.5-2

**Design Basis Missiles**

<table>
<thead>
<tr>
<th>Wind</th>
<th>Missile Type</th>
<th>Dimensions</th>
<th>Mass kg (lb)</th>
<th>Max. Horizontal Strike Velocity m/sec (ft/s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design Basis Tornado (1)</td>
<td>Schedule 40 Pipe</td>
<td>Ф 0.168 m × 4.58 m long (Ф 6.625 in × 15 ft long)</td>
<td>130 (287)</td>
<td>41 (135)</td>
</tr>
<tr>
<td></td>
<td>Automobile</td>
<td>5 m × 2 m × 1.3 m (16.4 ft × 6.6 ft × 4.3 ft)</td>
<td>1,810 (4,000)</td>
<td>41 (135)</td>
</tr>
<tr>
<td></td>
<td>Solid Steel Sphere</td>
<td>Ф 2.54 cm (Ф 1 in.)</td>
<td>0.0669 (0.147)</td>
<td>8 (26)</td>
</tr>
<tr>
<td>Design Basis Hurricane (2)</td>
<td>Schedule 40 Pipe</td>
<td>Ф 0.168 m × 4.58 m long (Ф 6.625 in × 15 ft long)</td>
<td>130 (287)</td>
<td>64.5 (212)</td>
</tr>
<tr>
<td></td>
<td>Automobile</td>
<td>5 m × 2 m × 1.3 m (16.4 ft × 6.6 ft × 4.3 ft)</td>
<td>1,810 (4,000)</td>
<td>80.2 (263)</td>
</tr>
<tr>
<td></td>
<td>Solid Steel Sphere</td>
<td>Ф 2.54 cm (Ф 1 in.)</td>
<td>0.0669 (0.147)</td>
<td>57.3 (188)</td>
</tr>
</tbody>
</table>

(1) The missile velocities in vertical direction are 67 percent of horizontal missile velocities. The automobile missile is considered to impact at all altitudes less than 10.06 m (33 ft) above all grade levels within 0.8 km (0.5 mi) of the plant structures.

(2) Missiles are based on the maximum wind velocity at elevation 10.06 m (33 ft) above ground. The design basis vertical missile velocity for all missiles is 26 m/s (58 mph).
Minimum Acceptable Barrier Thickness Requirements for Local Damage Prediction against Missiles Generated by Natural Phenomena

<table>
<thead>
<tr>
<th>Concrete Strength kg/cm² (psi)</th>
<th>Wall Thickness cm (in.)</th>
<th>Roof Thickness cm (in.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>350 (5,000)</td>
<td>51.1 (20.2)</td>
<td>43.7 (12.7)</td>
</tr>
<tr>
<td>420 (6,000)</td>
<td>50.3 (19.4)</td>
<td>42.5 (12.4)</td>
</tr>
</tbody>
</table>
Table 3.5-4 (1 of 2)

Essential Systems and Components Outside Reactor Containment Building to Be Protected from Externally Generated Missiles

<table>
<thead>
<tr>
<th>Protected Components</th>
<th>Missile Barrier</th>
</tr>
</thead>
<tbody>
<tr>
<td>Chemical and Volume Control System</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Charging pump mini-flow heat exchanger</td>
<td></td>
</tr>
<tr>
<td>Volume control tank</td>
<td></td>
</tr>
<tr>
<td>Charging pump (auxiliary charging pump)</td>
<td></td>
</tr>
<tr>
<td>Boric acid storage tank</td>
<td></td>
</tr>
<tr>
<td>Safety-related pipes and valves</td>
<td></td>
</tr>
<tr>
<td>Class 1E electric systems, including on-site safety-related portions of the Emergency Diesel Generator System necessary to provide emergency electric power to the other systems identified in this table.</td>
<td>Auxiliary Building and EDG Building</td>
</tr>
<tr>
<td>Spent Fuel Pool Cooling and Cleanup System</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Spent Fuel Pool Heat Exchanger</td>
<td></td>
</tr>
<tr>
<td>Spent Fuel Pool Cooling Pump</td>
<td></td>
</tr>
<tr>
<td>Safety-related Pipes and Valves</td>
<td></td>
</tr>
<tr>
<td>Main Steam System</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>MSIVs and pipe between MSIVs and containment</td>
<td></td>
</tr>
<tr>
<td>Shutdown Cooling System</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Shutdown Cooling Pump and Heat Exchanger</td>
<td></td>
</tr>
<tr>
<td>Safety-related Pipes and Valves</td>
<td></td>
</tr>
<tr>
<td>Essential Service Water System</td>
<td>ESW Building</td>
</tr>
<tr>
<td>Essential Service Water Pump</td>
<td></td>
</tr>
<tr>
<td>Safety-related Pipes and Valves</td>
<td></td>
</tr>
<tr>
<td>Control Room HVAC System</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>AHU, ACU</td>
<td></td>
</tr>
<tr>
<td>Control, Isolation and Smoke Damper</td>
<td></td>
</tr>
</tbody>
</table>

The BAST located in Yard is designed to withstand externally generated missiles.
# Table 3.5-4 (2 of 2)

<table>
<thead>
<tr>
<th>Protected Components</th>
<th>Missile Barrier</th>
</tr>
</thead>
<tbody>
<tr>
<td>Component Cooling Water System</td>
<td>Component Cooling Water Heat Exchanger Building and Auxiliary Building</td>
</tr>
<tr>
<td>Component Cooling Water Heat Exchanger</td>
<td>Component Cooling Water Heat Exchanger Building and Auxiliary Building</td>
</tr>
<tr>
<td>Component Cooling Water Pump</td>
<td>Component Cooling Water Heat Exchanger Building and Auxiliary Building</td>
</tr>
<tr>
<td>Component Cooling Water Makeup Pump</td>
<td>Component Cooling Water Heat Exchanger Building and Auxiliary Building</td>
</tr>
<tr>
<td>Safety-related Pipes and Valves</td>
<td>Component Cooling Water Heat Exchanger Building and Auxiliary Building</td>
</tr>
<tr>
<td>Auxiliary Feedwater System</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Auxiliary Feedwater Pump</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Auxiliary Feedwater Storage Tank</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Safety-related Pipes and Valves</td>
<td>Auxiliary Building</td>
</tr>
<tr>
<td>Liquid Waste Management System</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Floor drain tank</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Equipment waste tank</td>
<td>Compound Building</td>
</tr>
<tr>
<td>RW-IIa component in R/O package (Refer to Table 11.2-6)</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Gaseous Waste Management System</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Header drain tank</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Charcoal guard</td>
<td>Compound Building</td>
</tr>
<tr>
<td>bed Charcoal delay</td>
<td>Compound Building</td>
</tr>
<tr>
<td>bed HEPA filter</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Waste gas dryer</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Solid Waste Management System</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Low-activity spent resin tank</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Spent resin long-term storage tank</td>
<td>Compound Building</td>
</tr>
<tr>
<td>Concentrate treatment system</td>
<td>Compound Building</td>
</tr>
</tbody>
</table>
### Table 3.5-5 (1 of 2)

#### Allowable ductility Ratios

<table>
<thead>
<tr>
<th>Component</th>
<th>Loading Categories</th>
<th>Ductility Ratio</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reinforced Concrete</td>
<td>Concrete structure subjected to pressure pulse caused by compartment pressurization or external explosion load</td>
<td>1.0 for a structure as a whole 3.0 for a localized area</td>
<td>Except the case for axial compression + flexure</td>
</tr>
<tr>
<td>Flexure (beam, wall, and slabs)</td>
<td>Ductility Ratio: $\frac{0.05}{\rho - \rho'} \leq 10$ Rotational Capacity, $\gamma_\theta$: $\gamma_\theta &lt; 0.0065 \frac{d}{c} &lt; 0.07$ rad</td>
<td>$\rho$: tension reinforcement ratio $\rho'$: compression reinforcement Ratio $c$: distance from compression face to the neutral axis at ultimate strength $d$: distance from compression face to the tensile reinforcement - For flexure to control the design, the load capacity of a structural element in shear shall be at least 20% greater than the load capacity in flexure, otherwise, the ductility ratios given in shear or axial compression + flexure shall be used.</td>
<td></td>
</tr>
<tr>
<td>Shear (Beams, wall, and slab) carried by</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>concrete alone</td>
<td>1.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>concrete + stirrups or bent bars</td>
<td>1.3</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
### Table 3.5-5 (2 of 2)

<table>
<thead>
<tr>
<th>Component</th>
<th>Loading Categories</th>
<th>Ductility Ratio</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reinforced Concrete (continued)</td>
<td>Axial Compression + Flexure (beam-column, walls, and slabs)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>When compression load &gt; $0.1 f'<em>c A_g$ or $1/3 F</em>{bc}$</td>
<td>1.0</td>
<td>$F_{bc}$: Force which would produce balanced condition</td>
<td></td>
</tr>
<tr>
<td>When compression load ≤ $0.1 f'<em>c A_g$ or $1/3 F</em>{bc}$</td>
<td>Follow remarks</td>
<td>Ductility ratio for flexure is used</td>
<td></td>
</tr>
<tr>
<td>Combined compression + Flexure</td>
<td>Follow remarks</td>
<td>Ductility ratio is calculated linearly between 1.0 (Compression controls) and ductility ratio for flexure (Flexure controls)</td>
<td></td>
</tr>
</tbody>
</table>
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Figure 3.5-1  Turbine Missile Strike Zone
3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

According to 10 CFR Part 50, Appendix A, GDC 4 (Reference 1), essential structures, systems, and components (SSCs) important to safety are required to be protected from postulated piping failures. The effects of postulated piping failures include dynamic and environmental effects. Dynamic effects include pipe whipping, jet impingement, subcompartment pressurization, and fluid system decompression. Environmental effects include but are not limited to spray wetting, temperature, humidity, radiological consequences, and localized flooding.

Section 3.6 describes the design bases and design measures that are used to provide reasonable assurance that the safe shutdown capability, containment integrity, and offsite dose consequence are maintained for postulated piping failures inside and outside the containment. The postulated piping failures considered in this section include circumferential breaks, longitudinal breaks, and cracks.

The criteria used to evaluate the effects of postulated piping failures are generally consistent with NRC guidelines including those in Standard Review Plans 3.6.1, 3.6.2, and 3.6.3 (References 2, 3, and 4), and Branch Technical Positions 3-3 and 3-4 (References 5 and 6).

Subsection 3.6.1 identifies the high-energy and moderate-energy piping system and provides design bases and measures to protect essential SSCs from the effect of postulated piping failures and to verify integrity and operability.

Subsection 3.6.2 describes the criteria used to define locations and configurations of pipe rupture, and analytical methods to define forcing functions.

Subsection 3.6.3 describes the leak-before-break (LBB) analysis for all designated piping systems where application of LBB methodology is justifiable based on regulatory criteria. LBB analysis is used to eliminate from the structural design bases the dynamic effects of double-ended guillotine breaks and equivalent longitudinal breaks for an applicable piping system. However, environmental effects resulting from non-mechanistic piping failures are still applicable, as described above.
Plant Design for Protection against Postulated Piping Failures in Fluid Systems Inside and Outside the Containment

Protection against postulated fluid system piping failures is provided to the extent that the following criteria are met:

a. The minimum performance capabilities of the engineered safety systems are not reduced below those required to protect against the postulated break.

b. The ability to shut down the reactor and mitigate the consequences of the postulated piping failure.

c. Containment integrity is maintained if the postulated piping failure is a loss-of-coolant accident (LOCA).

d. Resultant radiological doses are below the guideline values of 10 CFR Part 100 (Reference 7).

3.6.1.1 Design Basis

Protection of essential equipment is achieved primarily by separation of redundant safe shutdown systems and by separation of high-energy pipelines from safe shutdown systems, which are required to be functional following specific pipe rupture events. This redundancy and separation result in a design that requires very few special protective features (such as pipe whip restraints and jet shields) to provide reasonable assurance of safe shutdown capability following a postulated high-energy line break (HELB).

Most systems and components outside the containment required for safe plant shutdown are located in the auxiliary building. The auxiliary building is divided by a structural wall that serves as a barrier between redundant trains of safe shutdown systems and components. Each half of the auxiliary building is compartmentalized to separate redundant safe shutdown components to the extent practical. High-energy piping systems located in the auxiliary building, which are not required to be functional for safe shutdown, are routed primarily in designated pipe tunnels or in the main steam valve houses (MSVHs) to provide separation from safe shutdown systems and components.

Systems and components inside the containment, which are required to be functional for safe plant shutdown, are protected from postulated pipe failure dynamic effects primarily by separation and barriers. The secondary shield wall serves as a barrier between the
reactor coolant loops and the containment liner. The steam generators (SGs) and pressurizer (PZR) are also enclosed in cavities that provide separation.

The systems that are required for safe shutdown or to support safe shutdown for a given pipe failure are described in Section 3.2 and listed in Table 3.2-1. High-energy and moderate-energy pipe failure locations are postulated as described in Subsection 3.6.2. A list of high- and moderate-energy fluid systems is provided in Table 3.6-1. Each postulated rupture location is evaluated for effect on safe shutdown systems and components required following the specific pipe failure event.

3.6.1.1.1 High-Energy Piping Systems

A high-energy pipe failure is postulated in branches or piping runs that are larger than 2.54 cm (1 in.) nominal diameter and operate with high-energy fluid during normal plant conditions.

Included in this category are fluid systems or portions of fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. Maximum operating temperature exceeds 93.3 °C (200 °F)

b. Maximum operating pressure exceeds 19.3 kg/cm² (275 psig)

In analyzing the effects of a high-energy pipe failure, the consequences of pipe whip, jet impingement, blast effects, flooding, subcompartment pressurization, and environmental conditions are considered.

3.6.1.1.2 Moderate-Energy Piping Systems

A moderate-energy pipe failure is postulated in branches or piping runs that are larger than 2.54 cm (1 in.) nominal diameter and operate with moderate-energy fluid during normal plant conditions.

Fluid piping systems that qualify as high energy for only short operational periods are considered moderate-energy systems if the fraction of the time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2 percent of the total time the system operates.
Included in this category are fluid systems or portions of fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. Maximum operating temperature is 93.3 °C (200 °F) or less

b. Maximum operating pressure is 19.3 kg/cm² (275 psig) or less

In analyzing the effects of a moderate-energy pipe failure, the consequences of flooding and environmental conditions are considered.

3.6.1.1.3 Failure Modes and Effects Analysis

An evaluation of postulated pipe break events is performed to identify the safety-related systems and components that provide protective actions required to mitigate, to acceptable limits, the consequences of the postulated pipe break event.

If the separation inherent in the plant design is shown to provide reasonable assurance of the functional capability of the safety systems required following a postulated pipe break event, no additional protective measures are required for that event. When necessary, additional protective measures are incorporated into the design to provide reasonable assurance of the functional capability of safety systems that are required following a postulated pipe break event.

The following criteria are used in the failure modes and effects analysis to establish the integrity of systems and components necessary for safe reactor shutdown and maintenance of the shutdown condition:

a. Offsite power is assumed to be unavailable if an automatic turbine generator trip or automatic reactor trip is a direct consequence of a postulated piping failure.

b. In addition to the postulated pipe failure and its accompanying effects, a single active component failure is assumed in the systems required to mitigate the consequences of the postulated piping failure except as noted in item d below.

c. Each high-energy or moderate-energy fluid system pipe failure is considered separately as a single postulated initiating event occurring during normal plant conditions.
d. Where a postulated piping failure is assumed in one of two redundant trains of a system that is required to operate during normal plant conditions as well as to shut down the reactor, single failures that prevent the functioning of the other train or trains of that system are not assumed, provided the system is (1) designed to seismic Category I standards, (2) powered from offsite and onsite sources, and (3) designed, constructed, operated, and inspected to quality assurance, testing, and in-service inspection standards appropriate for nuclear safety systems.

e. All available systems and components, including non-seismic Category I and those actuated by operator actions, may be used to mitigate the consequences of a postulated piping failure. In judging the availability of such systems and components, account is taken of the postulated failure and its direct consequences, such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences.

f. For a postulated pipe failure, the escape of steam, water, and heat from structures enclosing the high-energy fluid containing piping does not preclude:

1) Accessibility to surrounding areas important to the safe control of reactor operations

2) Habitability of the main control room (MCR)

3.6.1.2 Safety Evaluation

By means of design features such as separation, barriers, pipe whip restraints and jet shields, all of which are described below, the effects of a pipe break would not damage essential systems to the extent that the design function is impaired or the necessary component operability is affected.

Typical measures used for protecting the essential SSCs are described below.

3.6.1.2.1 Protection Measures

The design to protect essential SSCs from the effect of postulated break is basically achieved by separation, physical barriers, or piping restraint protection.
3.6.1.2.1.1 Separation

The plant arrangement provides separation to the extent practical between redundant safety systems in order to prevent loss of safety function as a result of hazards different from those for which the system is required to function, as well as for the specific event for which the system is required to be functional. Separation between redundant safety systems and associated auxiliary supporting features is the basic protective measure.

In general, layout of the facility followed a multistep process to provide reasonable assurance of adequate separation.

a. Safety-related systems are located away from most high-energy piping.

b. Redundant safety systems and subsystems are located in separate compartments.

c. As necessary, components are enclosed to maintain the redundancy required for the systems that must function as a consequence of piping failure events.

3.6.1.2.1.2 Barriers and Shields

Protection requirements are met through the protection afforded by the walls, floors, and columns in many cases. Where adequate protection does not exist due to separation, additional barriers, deflectors, or shields are provided as necessary to meet the functional protection requirements. Where compartments, barriers, and structures are required to provide the necessary protection, they are designed to withstand the effects of the postulated failure concurrent with an earthquake event.

Structures separating a high-energy line from an essential component are designed to withstand the consequences of a pipe break including associated pipe whip, jet impingement and subcompartment pressurization.

Structures separating a high-energy line from an essential component are designed to withstand the consequences of pipe break in high-energy line that produces the greatest effect at the structure.

3.6.1.2.1.3 Piping Restraints

Where adequate protection does not exist due to separation, barriers, or shields, piping restraints are provided as necessary to meet the functional protection requirements.
Restraints are not provided when it can be shown that the postulated pipe breaks would not cause unacceptable damage to essential systems or components.

The design criteria for pipe whip restraints are given in Subsection 3.6.2.4.

3.6.1.2.2 Specific Protection Consideration

The design criteria define acceptable types of isolation for safety-related elements and for high-energy lines from similar elements of the redundant train. Separation is accomplished by:

a. Routing the redundant trains through separate compartments

b. Physically separating the redundant trains by a specified minimum distance

c. Separating the redundant trains by structural barriers

The design criteria provide reasonable assurance that a postulated failure of a high-energy line or a safety-related element cannot take more than one safety-related train out of service. The failure of a component or subsystem of one train may cause failure of another portion of the same train; for example, a Division II high-energy pipe may cause failure of a Division II component electrical tray but not failure of any Division I component. The capability to safely shut down the plant under such a failure would therefore not be affected. Subsection 1.2.14.2 describes the detailed divisional separation of the auxiliary building.

Given the separation criteria above and the pipe break criteria in Subsection 3.6.2.1, the effects of high-energy pipe breaks are not analyzed where it is determined that all essential SSCs are sufficiently physically remote from a postulated break in that piping run.

The potential effects of flooding as a consequence of high- or moderate-energy pipe failure through-wall cracks (as defined in Subsection 3.6.2.1), are evaluated to provide reasonable assurance that the operability of safety-related equipment required for safe shutdown is not impaired. An analysis of the potential effects of flooding is described in Section 3.4.

Subcompartment pressurization analysis is performed to determine pressure loadings on building structures. Environmental pressure and temperature analysis defines pressure and temperature conditions for qualification of mechanical and electrical equipment. The pressure transients and environmental conditions for specified subcompartments are calculated using the code, and the results are reflected in the design of the subcompartments.
and venting paths. The subcompartment pressurization analysis is described in Subsection 6.2.1.2.

The potential environmental effects of steam on essential systems are described in Section 3.11. In general, because of the protective measures of redundancy and separation between systems and trains, the consequential effect of the transport of steam is not sufficient to impair the ability of the essential system to shut down the plant or mitigate the consequences of the given accident of interest.

There are no high-energy lines in the vicinity of the control room. As such, there are no effects on the habitability of the MCR by pipe break either from pipe whip, jet impingement, or transport of steam. Refer to Section 6.4 for a description of control room habitability systems.

3.6.1.3 Postulated Failures Associated with Site-Specific Piping

The COL applicant is to identify the site-specific safety-related SSCs that are located near high-energy and moderate-energy piping systems and are susceptible to consequences of high- and moderate-energy piping failures (COL 3.6(1)). The COL applicant is to provide a list of site-specific high-energy piping systems including layout drawings and protection features and the failure modes and effects analysis for safe shutdown due to the postulated HELB (COL 3.6(2)).

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section describes the design bases for locating breaks and cracks in piping inside and outside containment, the procedure used to define the thrust at the break location, the jet impingement loading criteria, and the dynamic response models.

3.6.2.1 Criteria Used to Define Rupture Locations and Configurations

3.6.2.1.1 General Requirements

Postulated pipe ruptures are considered in all plant piping systems and the associated potential for damage to required systems and components is evaluated on the basis of the energy in the system. System piping is classified as high-energy or moderate-energy, and postulated ruptures are classified as circumferential breaks, longitudinal breaks, or through-
wall cracks. Each postulated rupture is considered separately as a single postulated initiating event.

For each postulated circumferential and longitudinal break, an evaluation is made of the effects of pipe whip, jet impingement, blast effects, subcompartment pressurization, environmental conditions, and flooding. For piping systems where LBB is applied (Subsection 3.6.3), dynamic effects of pipe breaks are not considered. If required to demonstrate safe plant shutdown, an internal fluid system load evaluation is performed on the effects of fluid forces on components within or bounding the fluid system.

For each postulated through-wall crack, an evaluation is made of the effects of environmental conditions and flooding. The effects of pipe breaks and/or through-wall cracks are included in the environmental qualification of safety-related electrical and mechanical equipment. Environmental qualification of safety-related equipment is described in Section 3.11. The evaluation of the required systems and components demonstrates that the protection requirements of Subsection 3.6.1 are met.

The pipe break analysis is considered for postulated high-energy and moderate-energy line failure. Considering the following information, the plant can be shut down safely and maintained in cold safe shutdown when postulated pipe failure occurs. A graded approach is taken to the scope of piping system as described in DCD Tier 2 subsection 14.3.2.3. The results are summarized in the pipe rupture analysis report (Reference 24).

a. Dynamic effect analysis from high-energy line break

1) Identification of break and crack locations in high-energy and moderate-energy piping

2) Scope of dynamic effect analysis (Class 1 piping inside containment and piping connected NSSS component.)

3) Identification of terminal end break and intermediate break (result of stress analysis including fatigue analysis according to BTP 3-4)

4) High-energy line markup P&ID and break point dimensional isometric drawing

5) Essential target and dynamic load from pipe break and protection features
6) Summary of subcompartment pressure and temperature analysis for a 1 ft$^2$
break on the main steam and main feed lines, within the pipe break exclusion
zone.

b. Environmental analysis of the high-energy and moderate-energy piping systems to
protect safety-related system from flooding and other adverse environmental
effects.

3.6.2.1.2 Postulated Rupture Descriptions

a. Circumferential break

A circumferential break is assumed to result in pipe severance with full separation
of the two severed pipe ends unless the extent of separation is limited by
consideration of physical means. The break plane area is assumed to be
perpendicular to the longitudinal axis of the pipe and is assumed to be the cross-
sectional flow area of the pipe at the break location. The break flow area,
discharge coefficient, and discharge correlation are substantiated analytically.

b. Longitudinal break

A longitudinal break is assumed to result in a split of the pipe wall along the pipe
longitudinal axis but without severance. The break plane area is assumed to be
parallel to the longitudinal axis of the pipe and equal to the cross-sectional flow
area of the pipe at the break location. The break is assumed to be circular in
shape or elliptical ($2D \times D/2$) with its long axis parallel to the longitudinal axis of
the pipe where $D$ is the effective inner diameter of the pipe. The discharge
coefficient and any other values used for the area or shape associated with a
longitudinal break are substantiated analytically.

c. Leakage crack

A leakage crack is assumed to be a crack through the pipe wall where the size of
the crack and corresponding flow rate are determined by analysis and a leak
detection system, as described in Subsection 3.6.3.

d. Through-wall crack
A through-wall crack is assumed to be a circular orifice through the pipe wall of cross-sectional flow area equal to the product of half of the inside pipe diameter and half of the pipe wall thickness.

3.6.2.1.3 Leak-Before-Break Applied Piping

An LBB evaluation has been performed for the reactor coolant system (RCS) main loop piping, surge line, shutdown cooling, and safety injection lines inside containment, which eliminates the dynamic effects of pipe break from the design basis. The evaluation is performed using the guidelines of NUREG-1061 (Reference 8), Vol. 3, and SRP 3.6.3, as described in Subsection 3.6.3.

3.6.2.1.4 Piping Not Applied to Leak-Before-Break

This section applies to all high- and moderate-energy piping other than that whose dynamic effects due to pipe breaks are eliminated from the design basis by LBB evaluation, as identified in Subsection 3.6.2.1.3.

3.6.2.1.4.1 Postulated Rupture Locations

3.6.2.1.4.1.1 Break Locations for High-Energy Fluid System Piping in Areas Other than Containment Penetration

Both circumferential and longitudinal breaks of high-energy piping systems are postulated to occur, but not concurrently, considering the following exceptions:

a. Circumferential breaks are not postulated in piping runs of a nominal diameter equal to or less than 2.54 cm (1 in.).

b. Longitudinal breaks are not postulated in piping runs of a nominal diameter less than 10.16 cm (4 in.).

c. Longitudinal breaks are not postulated at terminal ends.

The postulated pipe break locations are determined in accordance with BTP 3-4, Part B, item A(iv). All pipes are identified and considered for the selection of break locations.

A terminal end is defined as an extremity of a piping run that connects to structures, components, or pipe anchor that acts as a rigid constraint to piping motion and thermal
expansion. Branch lines connected to main piping run are considered as terminal end in accordance with BTP 3-4.

**ASME Section III (Reference 9), Class 1 Piping**

With the exceptions of the portions of piping identified in Subsection 3.6.2.1.3, breaks in ASME Section III, Class 1 piping are postulated at the following locations in each piping and branch run:

a. At terminal end

b. At intermediate locations where $U$ exceeds 0.1 or $S$ from Equation (10) plus Equation (12) or (13) exceeds $2.4 S_m$

Where, as defined in ASME Section III, Division 1, NB-3653:

$$S = \text{primary-plus-secondary stress-intensity range under the combination of loadings for which either Level A or Level B service limits have been specified, as calculated from Equations (10), (12), and (13)}$$

$$S_m = \text{allowable stress-intensity value}$$

$$U = \text{cumulative usage factor}$$

As a result of piping reanalysis due to differences between the design configuration and the as-built configuration, the highest stress or cumulative usage factor locations may be shifted. However, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:

a. The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

b. A change is required in pipe parameters such as major differences in pipe size, wall thickness, and routing.
ASME Section III, Class 2, Class 3, or Seismically Analyzed ASME B31.1 (Reference 10)

Piping

With the exceptions of the portions of piping identified in Subsection 3.6.2.1.3, breaks in ASME Section III, Class 2&3 piping are postulated at the following locations in each piping and branch run:

a. At terminal end

b. At intermediate locations selected by one of the following criteria:

1) At each pipe fitting (e.g., elbow, tee, cross, flange, nonstandard fitting), welded attachment, and valve or where the piping contains no fittings, welded attachments, or valves at one location at each extreme of the piping adjacent to the protective structure.

2) At intermediate locations where the stress $S$ exceeds $0.8 (X + Y)$

Where, as defined in ASME Section III, Division 1, NC/ND-3653

$S = \text{stresses under the combination of loadings for which either level A or level B stress limits have been specified, as calculated from the sum of Equations (9) and (10)}$

$X = \text{Equation (9) service level B allowable stress}$

$Y = \text{Equation (10) allowable stress}$

As a result of piping reanalysis due to differences between the design configuration and the as-built configuration, the highest stress locations may be shifted; however, the initially determined intermediate break locations may be used unless a redesign of the piping resulting in a change in pipe parameters (diameter, wall thickness, routing) is required, or the dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

Non-Seismically Analyzed ASME B31.1 Piping

Safety class systems and piping are seismically analyzed. Most non-safety systems and piping are required to be non-seismically analyzed. Therefore, rules are isolation and
separation in order to avoid impact to safety-related systems from non-safety-related piping failure effects caused by an earthquake. In cases where it is not possible or practical to isolate the seismic piping, adjacent non-seismic piping is analyzed according to seismic Category II criteria.

For non-seismic piping attached to seismic piping, the dynamic effects of the non-seismic piping are simulated in the modeling of the seismic piping. The attached non-seismic piping up to the analyzed/unanalyzed boundary is designed not to cause a failure of the seismic piping during a seismic event as described in Subsection 3.12.3.7.

Breaks are postulated at the following locations in each non-safety-related, ASME B31.1 piping network that is not seismically analyzed.

   a. At terminal ends of the pressurized portions of the network
   b. At each intermediate location of potential high stress or fatigue, such as pipe fittings, valves, flanges, and welded-on attachments

3.6.2.1.4.1.2 Crack Locations

Through-wall Cracks

Through-wall cracks are postulated in all high-energy and moderate-energy piping systems having a nominal diameter greater than 2.54 cm (1 in.), except that through-wall cracks are not postulated at locations where:

   a. For Class 1 piping, the calculated value of stress range by equation (10) in ASME Section III, Division 1, NB-3653 is less than 1.2 S_m.
   b. For Class 2, Class 3, or seismically analyzed ASME B31.1 piping, the calculated stress by the sum of equations(9) and (10) in NC/ND-3653 is less than 0.4 times the sum of the stress limits given in NC/ND-3653.
   c. For the containment penetration area, the requirements of Subsection 3.6.2.1.4.1.3 are satisfied.
   d. For moderate-energy piping near high-energy piping where a break in high-energy piping is postulated that results in more limiting environmental conditions, the through-wall crack in the moderate-energy piping is not postulated.
e. Through-wall cracks, instead of breaks, are postulated for the piping classified as moderate energy as defined in subsection 3.6.1.1.2 due to its short operational period in the high-energy condition.

Leakage Cracks

A leakage crack is postulated in place of a circumferential break, longitudinal break, or through-wall crack if justified by an analysis performed on the pipeline in accordance with the requirements described in Subsection 3.6.3.

3.6.2.1.4.1.3 Fluid System Piping in Containment Penetration Areas

3.6.2.1.4.1.3.1 High-Energy Piping

ASME Section III, Class 2 piping is used in containment penetration areas. Breaks or cracks are not postulated in those portions of piping between the containment wall and the inboard or outboard isolation valves, because they are designed with beak exclusion by meeting the design criteria of the ASME Section III, NE-1120 and the following additional requirements:

a. The following design stress and fatigue limits are not exceeded:

1) The maximum stress as calculated by the sum of Equations (9) and (10) in NC-3653, ASME Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system’s Design Specification (i.e., sustained loads, occasional loads, and thermal expansion), excluding earthquake loads, does not exceed 0.8 \((1.8 S_h + S_A)\). The \(S_h\) and \(S_A\) are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in NC-3600 of ASME Section III.

2) The maximum stress, as calculated by Equation (9) in NC-3653, ASME Section III, under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping, does not exceed the lesser of 2.25 \(S_h\) and 1.8 \(S_y\).

3) Primary loads include those that are deflection limited by whip restraints. The exceptions permitted in BTP 3-4, Part B, Item A (ii)(1)(c) may also be applied provided that when the piping between the outboard isolation valve
and the restraint is constructed in accordance with Power Piping Code ANSI B31.1 (see ASB 3-1 B.2.c.[4]), the piping is either of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds are fully radiographed.

b. Welded attachments for pipe supports or other purposes to these portions of piping are avoided except where detailed stress analyses or tests are performed to demonstrate conformance with the limits of Item a above.

c. The number of circumferential and longitudinal piping welds and branch connections is minimized. Specific access provisions are made to permit in-service volumetric examination of the longitudinal and circumferential welds.

d. The length of these portions of piping is reduced to the minimum length practical.

e. The design of pipe anchors or restraints (e.g., connections to containment penetrations, pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate conformance with the limits in Item 1 above.

f. A 100 percent volumetric in-service examination of all pipe welds is conducted during each inspection interval as defined in IWA-2400, ASME Section XI (Reference 11).

Since guard pipes are not used, the guard pipe design requirements addressed in BTP 3-4, Part B, Item A(ii)(6) are excluded from the additional requirements. Pipe break exclusion design expands to the auxiliary building wall anchor beyond the isolation valve.

Portions of system piping that are designed using break exclusion criteria are as follows:

a. The main steam piping from the containment penetration to the auxiliary building MSVH anchor wall, which is downstream of the main steam isolation valve.

b. The main feedwater piping from the containment penetration to the auxiliary building MSVH anchor wall, which is upstream of the isolation valve.
c. The steam generator blowdown piping from the containment penetration outboard weld to the auxiliary building wall, which is downstream of the isolation valve.

d. The startup feedwater piping from the containment penetration outboard weld to the auxiliary building MSVH anchor wall, which is upstream of the isolation valve.

Even though portions of the main steam and feedwater lines meet the break exclusion requirements of Subsection 3.6.2.1.4.1.3.1, they are separated from essential equipment.

Essential equipment is not concentrated in the break exclusion zone. Essential equipment is protected from the environmental effects of an assumed nonmechanistic longitudinal break of the main steam and feedwater lines. Each assumed nonmechanistic longitudinal break has a cross-sectional area of at least 1 ft² and is postulated to occur at a location that has the greatest effect on essential equipment.

3.6.2.1.4.1.3.2 Moderate-Energy Piping

For moderate-energy fluid systems, through-wall cracks are not postulated in those portions of piping from the containment wall to and including the inboard or outboard isolation valves provided that they meet the requirements of the ASME Section III, NE-1120, and the stresses calculated by the sum of Equations (9) and (10) of the ASME Section III, NC-3653, do not exceed 0.4 times the sum of the stress limits given in ASME Section III, NC-3653.

3.6.2.1.4.2 Postulated Rupture Configurations

a. Break configurations

Where break locations are postulated without the benefit of a stress calculation, breaks are assumed to occur at the piping welds to each fitting, valve, or welded attachment. If detailed stress analyses or tests are performed, break locations are selected as specified in Subsection 3.6.2.1.4.1.1.

Circumferential breaks are postulated in fluid system piping and branch runs as specified in Subsection 3.6.2.1.4.1.1. Instrument lines with tubing size of 2.54 cm (1 in.) and less nominal pipe are designed to meet the provisions of NRC RG 1.11 (Reference 12).

Circumferential breaks are assumed to result in pipe severance and separation amounting to at least an one-diameter lateral displacement of the ruptured piping
sections unless physically limited by piping restraints, structural members, or piping stiffness. The break plane area \((A_e)\) is assumed perpendicular to the longitudinal axis of the pipe, and is assumed to be the cross-sectional flow area of the pipe at the break location. The break flow area, discharge coefficient, and discharge correlation are substantiated analytically.

Longitudinal breaks in fluid system piping and branch runs are postulated as specified in Subsection 3.6.2.1.4.1.1.

A longitudinal break results in an axial split without severance. The split is assumed to be oriented at any point about the circumference of the pipe, or alternatively at the point of highest stress as justified by detailed stress analyses. The break plane area \((A_e)\) is assumed parallel to the longitudinal axis of the pipe and equal to the cross-sectional flow area of the pipe at the break location. The break is assumed to be circular in shape or elliptical \((2D \times D/2)\) with its long axis parallel to the axis of the pipe.

b. Crack configurations

Through-wall cracks are postulated at those axial locations specified in Subsection 3.6.2.1.4.1.2.

For high and moderate-energy piping including non-seismically analyzed ASME B31.1 piping, through-wall cracks for determining environmental conditions described in Table 3.11-2 are postulated to be in those axial and circumferential locations that result in the most severe environmental consequences. The flow from the crack is assumed to wet all unprotected components within the compartment with consequent flooding in the compartment and communicating compartments.

Fluid flow from a leakage crack is based on a circular orifice with a cross-sectional area equal to that of a rectangle of one-half the pipe inside diameter in length and one-half the pipe wall thickness in width.

3.6.2.1.5 Details of Containment Penetrations

Details of containment penetrations are described in Subsections 3.8.1 and 3.8.2.
3.6.2.2 Guard Pipe Assembly Design Criteria

Guard pipes are not used in any containment penetrations of high-energy piping.

3.6.2.3 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.3.1 Leak-Before-Break Applied Piping

There are no forcing functions or response models for the piping qualified for LBB.

3.6.2.3.2 Analytical Methods to Define Forcing Functions and Response Models for Piping Not Applied to Leak-Before-Break

This subsection applies to all high-energy piping other than that whose dynamic effects due to pipe breaks are eliminated from the design basis by LBB evaluation.

3.6.2.3.2.1 Determination of Pipe Thrust and Jet Loads

3.6.2.3.2.1.1 Dynamic Force of the Fluid Jet Discharge

The dynamic force of the fluid jet discharge from either a postulated circumferential or longitudinal break is based on a circular break area equal to the cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure multiplied by an analytically determined thrust coefficient, as determined for a circumferential break at the same location.

Flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.

Piping movement is assumed to occur in the direction of the jet reaction, unless limited by structural members or piping restraints.

Blowdown thrust force ($F_{\text{thrust}}$) is determined by multiplication of three parameters (i.e., appropriately calculated thrust coefficient considering the fluid condition, piping internal pressure, cross-sectional area at the break location). The initial conditions of piping pressurized during operation at power are the greater of the contained energy at 102 percent power or hot standby in accordance with SRP 3.6.2, Section III.4.A. For the realistic
calculation, $P_a A$ acting on the outside of the pipe may be subtracted from the thrust force acting on the inside of the pipe.

$$F_{\text{thrust}} = (C_T \ P_o - P_a) \ A$$

At first, initial, intermediate, and final thrust coefficients are calculated.

$$F_{\text{ini}} = 1.0 \ P_o \ A, \text{ or } F_{\text{ini}} = (1.0 \ P_o - P_a) \ A$$

$$F_{\text{inter}} = C_{TI} \ P_o \ A, \text{ or } F_{\text{inter}} = (C_{TI} \ P_o - P_a) \ A$$

$$F_{\text{final}} = C_{TS} \ P_o \ A, \text{ or } F_{\text{final}} = (C_{TS} \ P_o - P_a) \ A$$

Where:

- $D =$ pipe inside diameter based on the average or nominal pipe wall thickness
- $P_o =$ initial fluid pressure in source or pipe
- $P_a =$ atmospheric pressure
- $A =$ pipe flow area
- $C_T =$ thrust coefficient
- $C_{TI} =$ intermediate thrust coefficient
- $C_{TS} =$ steady-state thrust coefficient

This methodology is based on the simplified methods described in ANSI/ANS 58.2 (Reference 13) and pipe thrust and jet loads (Reference 14).

3.6.2.3.2.1.2 Evaluation of Jet Impingement Effects

Jet impingement force calculations are performed only if structures or components are adjacent to postulated HELBs, and it cannot be demonstrated that failure of the structure or component would not adversely affect safe shutdown capability. Details for jet impingement effects are described in Section 3.6.2.4.2.
Method of Dynamic Analysis of Unrestricted Pipes

The impact velocity and kinetic energy of unrestricted pipes are calculated on the basis of the assumption that the segments at each side of the break act as rigid-plastic cantilever beams subject to piecewise constant blowdown forces. The hinge location is fixed either at the nearest restraint or at a point determined by the requirement that the shear at an interior plastic hinge is zero. The kinetic energy of an accelerating cantilever segment is equal to the difference between the work done by the blowdown force and that done on the plastic hinge.

\[ KE = (TL_h - M_p) \theta \]

Where:

- **KE** = impact energy
- **T** = thrust force
- **L_h** = length of hinge
- **\( \theta \)** = angle of hinged pipe
- **M_p** = plastic section modulus of pipe

The impact velocity \( V_1 \) is found from the expression for the kinetic energy:

\[ V_1^2 = \frac{2gKE}{W_{eq}} \]

Where:

- **W_{eq}** = equivalent weight of the cantilever

For a straight run of pipe rotating about a plastic hinge, the zone of influence of the whipping pipe accounts for an increasing length due to a traveling hinge point caused by strain hardening effects. The impact energy of unrestrained pipe into a barrier (e.g., the divisional wall) is governed by the vector component of its velocity at impact that is perpendicular to the barrier. Impact of small piping into building structures conservatively assumes that all of the impact energy is imparted to the barrier with no dissipation due to local crushing deformation of the pipe.
Bearing area of impact on building structure is generally elliptical, but is treated as a circle of equivalent area, with dimensions based on experimental data for pipe crush behavior. As the impact load is greatest on the periphery of the ellipse, this yields a conservative force distribution into the barrier. Long-term loading on the barrier subsequent to impact due to system blowdown and continued deceleration of remaining pipe (beyond the impact zone) is accounted for in addition to the initial impulsive loading.

Unless a detailed analysis is performed to evaluate the response of an impacted pipe, the impacted pipe of smaller size and wall thickness is assumed to be failed at the point of initial contact. The impacted pipe of larger or equal size with thinner wall thickness is assumed to develop a through-wall crack. If the impacted pipe has both nominal size and wall thickness larger than or equal to that of the whipping pipe, its pressure boundary integrity is assumed to be undamaged.

3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.4.1 Pipe Whip Restraints

This section applies to pipe whip restraints for all piping other than that whose dynamic effects due to pipe breaks are eliminated from the design basis by LBB evaluation.

When required, pipe whip restraints are provided to protect the safety-related component against the effects of pipe whipping during postulated pipe break. The design of pipe whip restraints is governed not only by the pipe break blowdown thrust, but also by functional requirements, deformation limitations, properties of whipping pipe, and the capacity of the support structure. The restraint is designed for the impact force induced by the maximum possible initial gap between the whip restraint and the process pipe.

The impact energy is usually too high for an elastic restraint system or support structure to absorb. Therefore, energy-absorbing restraints are designed using the energy balance approach (impact energy + external work = internal energy of pipe restraint system).

3.6.2.4.1.1 Pipe Whip Restraint Components

Pipe whip restraints typically consist of the components listed below. The typical shape of a pipe whip restraint is shown in Figure 3.6-1.

a. Energy-absorbing members
Members that absorb energy by significant plastic deformations under the influence of impacting pipes (pipe whip). Crushable honeycomb material is not used in pipe whip restraint systems.

b. Non-energy-absorbing members

Components that form a direct link between the pipe and the structure.

c. Structural attachments

Fasteners that provide the method of attaching connecting members to the structure (e.g., welds, bolts).

d. Support structure

Steel and concrete support structures that ultimately carry the restraint load. Design criteria are specified in Subsections 3.8.3 and 3.8.4.

3.6.2.4.1.2 Methods for the Dynamic Analysis of Pipe Whip

A clearance between a pipe whip restraint and pipe is provided for thermal movement of pipe during normal operation and used as the maximum possible initial gap in the dynamic analysis. If a break occurs, the restraints or anchors nearest the break point prevent unlimited movement of the pipe at the point of break. In the absence of analytical justification, a dynamic load factor (DLF) of 2.0 is applied in determining a restraint loading to consider dynamic nature of the piping thrust load. Elasto-plastic pipe and whip restraint material properties may be considered as applicable. The effect of rapid strain rate of material properties is considered in accordance with ANSI/ANS 58.2. A 10 percent increase in yield strength is used to account for strain rate effects.

In general, the loading that may result from a break of piping is determined using either a dynamic blowdown or a conservative static blowdown analysis. The two methods that are used for analyzing the interaction effects between a whipping pipe and a restraint are energy balance method and equivalent static method.

The energy balance method is based on the principle of conservation of energy. The kinetic energy of the whipping pipe generated during the first quarter-cycle of movement is assumed to be converted into equivalent strain energy, which is distributed to the pipe or the whip restraint.
An equivalent static analysis model is used for rigid rupture restraints. In order to obtain the design load for a rigid restraint, the following equation is used:

\[ F = 2 \times 1.1 \times F_B = 2.2F_B \]

Where:

- \( F \) = design load
- \( F_B \) = maximum blowdown force

The DLF is taken as 2.0 and rebound effects are accounted for by a factor of 1.1.

3.6.2.4.1.3 **Design Loads**

Restraint design loads are established using the criteria delineated in Subsection 3.6.2.4.1.2.

3.6.2.4.1.4 **Allowable Stresses**

The strain of energy-absorbing members is limited to 50 percent of the ASTM-specified minimum elongation. The yield stress for energy-absorbing members is equal to using a dynamic yield stress (\( F_{yd} \)) that is 1.1 times the static yield stress (\( F_{ys} \)) specified in ASTM.

The allowable stresses for non-energy-absorbing members, structural attachment, and support steel structure are specified in AISC N690. In evaluating allowable stresses, dynamic yield stress (\( F_{yd} \)) is used for dynamic loads and static yield stress (\( F_{ys} \)) for static loads.

3.6.2.4.1.5 **Design Criteria**

The unique features in the design of pipe whip restraint components relative to the structural steel design are geared to the loads used and the allowable stresses. These are as follows:

a. Energy-absorbing members are designed for the restraint reaction and the corresponding deflection established according to size and material of pipe, and the blowdown force delineated in Subsection 3.6.2.4.

b. Non-energy-absorbing members, structural attachments, and the support structure are designed for the design load delineated in Subsection 3.6.2.4.1.3.
Material

The materials used for pipe whip restraint components are as follows:

a. For energy-absorbing member: ASTM A193, Grade B7
b. For other members: ASTM A500, Grade B, ASTM A572, Grade 50, ASTM A36

Jet Impingement on Essential Piping and Components

The criteria and procedures for evaluating jet impingement on essential piping and components consider the guidance of SRP 3.6.2 Revision 3 (Reference 3), which identifies potential concerns with the guidance of ANSI/ANS 58.2. In particular, the NRC has questioned jet spreading and pressure distribution within the jet as described in the standard. ANSI/ANS 58.2 includes the option to use NUREG/CR-2913 (Reference 15), which is not explicitly questioned in SRP 3.6.2. Jet impingement in the APR1400 has been considered separately for the different fluid conditions at each HELB location. Specifically, single phase steam, saturated (i.e., two phase steam/water), and liquid jets are addressed through different methodologies that consider:

a. Jet range, shape, and direction (i.e., ZOI)

b. Jet blowdown pressure distribution

c. Jet impingement force, including thrust coefficient

Single-phase liquid jets are assumed not to expand since water is almost incompressible and is issuing from a straight pipe, so the cross-sections of their ZOIs are the same as those of the breaks themselves. The penetration distance for a liquid jet is assumed to extend infinitely until it impinges on a target. Because the distances from potential HELBs to nearby surfaces are relatively short, gravitational effects are not considered. A thrust coefficient of 2.0 is applied.

Two-phase jets are analyzed according to the methodology in NUREG/CR-2913. The jet expansion is determined according to the numerical analyses documented in Chapter 2 and Appendix A of NUREG/CR-2913 and is not dependent on a generalized expansion model, which was the focus of the NRC concern with ANSI/ANS 58.2. Two-phase jets are analyzed up to the penetration distance specified of 10 L/D, which is supported by the experimental foundation of NUREG/CR-2913.
For steam jets, axisymmetric CFD analyses using the ANSYS CFX Version 17.0 code were performed to determine the characteristics of free jets (i.e., unimpinged) for three scenarios that are representative of the range of steam conditions and pipe diameters in the APR1400. The CFD simulations represent the steady-state conditions, assuming the upstream source pressure and temperature remain fixed. The resulting pressure distribution contours display a characteristic Mach disk, within which the pressure is essentially constant at the stagnation pressure at the outlet. Outside of the Mach disk, the pressure drops off very rapidly, except for a downstream barrel shock. The CFD analyses were used to determine:

a. A conservatively large ZOI that bounds any of the thermodynamic conditions for steam jets in the APR1400. The initial spreading angle is broader than that in ANSI/ANS 58.2 (i.e., the Modified Moody Model). Although the jet persists beyond a distance 25 L/D, the pressures farther out are localized and fairly low. Due to the limited distances inside the APR1400 plant, a penetration limit of 25 L/D is considered to bound the range of impingement effects.

b. A conservative static loading methodology, which bounds the thermodynamic conditions found in the APR1400 to apply to targets inside the ZOI. This involves characterizing the pressure internal to the jet as it expands. Although steam jet pressure drops off rapidly with distance beyond the Mach disk, no credit is taken for this decrease as the jet expands and pressure decreases. However, the total force cannot exceed that at the break exit. In addition, for targets smaller than the jet ZOI, the force exerted on the target will be limited due to the smaller area of intersection between the target and the jet. A thrust coefficient of 1.26 is applied for steam jets, which is consistent with guidance in the SRP 3.6.2.

The qualification of essential piping and components includes jet impingement in Service Level D. Concurrent loads include deadweight, thermal, seismic, pressure, and concurrent effects associated with the postulated break.

Where necessary, protection from jets is provided by using jet shields.

3.6.2.4.2.1 Potential for Dynamic Amplification and Jet Resonance

SRP 3.6.2 identifies a concern regarding the potential for jet load amplification associated with formation of unsteadiness in free jets, especially supersonic jets, that propagates in the shear layer to induce time-varying oscillatory loads on obstacles in the flow path. The concern is that synchronization of transient waves with the shear layer vortices emanating
from the jet break can lead to significant amplification of the jet pressures and forces (a form of resonance). Should the dynamic response of the neighboring structure also synchronize with the jet loading time scales, further amplification of the loading can occur as the result of formation of a feedback loop. When the impingement surface is within 10 diameters of the jet opening, and when resonance within the jet occurs, significant amplification of impingement loads might result.

Dynamic amplification and resonance does not occur for a HELB at potential locations in the APR1400. This effect occurs where engineered nozzles discharge single phase, high speed jets. The jet issuing from a pipe break is under-expanded. Although there may be supersonic regions downstream of the exit, the jet conditions are not consistent with those needed for dynamic amplification. For the APR1400, multiple physical characteristics of HELBs prevent occurrence of a resonance. These include self-damping effects of a two-phase jet (which is not relevant to single phase jets where resonance has been seen), lack of perpendicular flat surfaces of sufficient size to establish a feedback loop, variation in jet discharge angle and distance that prevent establishing a stable feedback loop, and irregularities in the contours of the broken pipe end and the impingement target that distort the outgoing jet and spread out reflected acoustic energy. The absence of HELB resonant effects is substantiated by a survey of experimental results. Therefore, resonant induced pressure loading is not a concern for jet impingement upon SSCs of the APR1400.

3.6.2.4.3 Implementation of Criteria Dealing with Special Features

Special features used for piping protection (e.g., pipe whip restraints) are described in Subsection 3.6.2.4. Special features such as augmented in-service examination are described in Subsection 3.6.2.1.4.1.3, Item f.

3.6.2.4.4 Blast Effects

SRP 3.6.2 identifies that the first significant fluid load on surrounding SSCs due to a HELB is possible to be induced by a blast wave. Although a spherically expanding blast wave is reasonably approximated to be a short-duration transient and analyzed independently of any subsequent jet formation, reflections and amplifications in enclosed areas of the plant may need to be evaluated. Blast waves are not considered in the ANSI/ANS 58.2 Standard for evaluating the dynamic effects associated with the postulated pipe rupture, and no regulatory guidance or acceptance criteria exist for this evaluation.
The formation of a blast wave and its propagation are complex, interactive phenomena with limited data available to characterize the shock loads due to a HELB. Computational Fluid Dynamics (CFD) modeling was performed for some of the most limiting steam break locations. HELBs in portions of systems with subcooled liquids form only weak, if any, shock waves because boiling fronts propagate at velocities that are orders of magnitude slower than the pressure waves in a liquid, so a liquid layer may remain superheated after a HELB for an extended period and not be able to participate in formation of a shock wave.

The CFD analysis approach was benchmarked against several experiments and analyses of similar conditions to verify its suitability.

CFD results showed that a moderate pressure shock wave formed rapidly, propagated away from the HELB location, and was subject to reflections that resulted in higher localized pressures at certain points on SSC surfaces. The effects of plant-specific geometry, such as proximity to a large structure, on surface pressure were identified. In particular, a shock wave issuing from a pipe that has just broken is directional, rather than spherically uniform, but the amount of energy that can participate in forming the shock wave is limited by the fluid that can be discharged before the shock initiates. In all cases, the shock wave pressure positive impulse was brief (a few milliseconds), which limits the impact upon SSCs in its path.

The results from the APR1400-specific CFD analysis were used to develop a simplified methodology that applies an experimentally-determined pressure vs. distance relationship found in the literature. The severity of the blast effects depends on the initial energy put into the shock wave, which is conservatively determined. The methodology then accounts for effects of plant geometry on reflection and dissipation to determine a conservative estimate of the surface pressure at a point on the surface of an SSC.

3.6.3 Leak-Before-Break Evaluation Procedure

This section describes LBB analysis for all applicable piping. LBB analysis is used to eliminate from the structural design basis the dynamic effects of double-ended guillotine breaks and equivalent longitudinal breaks for an applicable piping system.

According to 10 CFR Part 50, Appendix A, GDC 4, the exclusion of dynamic effects due to the pipe rupture from the design basis is allowed when the analyses demonstrate that the probability of pipe rupture is extremely low under conditions satisfied by the design basis
such as normal operation, thermal transients, and seismic events in the process of approval and review.

This section describes how the piping system meets the LBB criteria in accordance with SRP 3.6.3 and demonstrates that the probability of pipe rupture is extremely low. The steps below are followed to carry out the LBB analyses.

a. Evaluate the potential failure mechanism

b. Perform the LBB analysis using piping evaluation diagram (PED)

The method of PEDs allows for the evaluation of the piping system in advance of the final piping analysis, incorporating LBB considerations into the piping design.

The COL applicant is to confirm that the bases for the LBB acceptance criteria are satisfied by the final as-built design and materials of the piping systems as site-specific evaluations, and is to provide the information including LBB evaluation report for the verification of LBB analyses (COL 3.6(3)).

3.6.3.1 Application of Leak-Before-Break

LBB is applied only to the following piping systems:

a. Reactor coolant loop (RCL) piping, hot leg (HL) and cold leg (CL)

b. Surge line (SL)

c. Direct vessel injection (DVI) line (main run inside containment)

d. Shutdown cooling (SC) line (main run inside containment)

3.6.3.2 Piping Design for Leak-Before-Break Application

The approach taken in the APR1400 design is to include LBB considerations in the piping design. One aspect of the LBB evaluation pursued for each selected piping system is the performance of a preliminary LBB evaluation prior to and independent of pipe routing. This evaluation is used to establish LBB acceptance criteria in terms of a range of normal operating and maximum design loads for each piping system designed for LBB. An LBB piping evaluation diagram is established, which is used to route, design and support the piping system within a range of design parameters. Site-specific evaluations will confirm
that the bases for the acceptance criteria are satisfied by the final as-built design and materials.

Piping systems designed for LBB meet the requirements defined by the LBB piping evaluation diagrams, which constitute the crack stability acceptance criteria. The LBB piping evaluation diagrams are based on a defined set of piping design parameters for each piping system.

3.6.3.2.1 Piping Design Requirements

Piping design parameters, including pipe size, base metal and weld material, and leak detection capability, form the basis for developing LBB acceptance criteria curves (i.e., the LBB piping evaluation diagrams, PEDs). To demonstrate that LBB acceptance criteria are met, the piping systems are designed such that the load at all locations determined in accordance with the selection criteria of Subsection 3.6.3.5.1 falls within the acceptance area on the piping evaluation diagram. A methodology for developing LBB piping evaluation diagrams is discussed in Subsection 3.6.3.5.5.

Piping design parameters used for developing the PEDs constitute piping design requirements for LBB for a specific pipeline. In addition to the criteria defined by the pipe specific PED, the piping systems, to which LBB is applied, meet the LBB applicability criteria discussed in Subsection 3.6.3.4.

3.6.3.2.2 Piping Design Procedure

The piping is routed, designed, and analyzed in accordance with the ASME Boiler and Pressure Vessel Code, and is evaluated for LBB using the criteria discussed in Subsection 3.6.3.3. As-calculated piping loads at all locations determined in accordance with the selection criteria of Subsection 3.6.3.5.1 are compared to the LBB PED which is based on the set of design parameters for that piping system.

3.6.3.2.3 Plant and Piping Design Conditions

3.6.3.2.3.1 Piping Design Parameters

In piping design, fluid system requirements normally drive the selection of specific piping parameters. For piping systems chosen for LBB evaluation, LBB considerations are integrated into the process of selecting those design parameters. Specifically, the design
parameters, for which LBB is considered, include pipe size (cross-section), pipe and weld materials, loads and piping system thermal flexibility.

The pipe and weld materials are chosen considering LBB requirements, along with system, stress and fatigue requirements. Within the limitations of fluid system and ASME Code requirements, pipe and weld materials are selected, which have good corrosion resistance, high yield and high toughness characteristics.

Piping system thermal flexibility is governed by the stress requirements of the ASME Code and the duty cycle of loadings. The piping systems are routed such that they are sufficiently flexible to thermally deflect without exceeding stress or fatigue limits and meet criteria for all load combinations associated with earthquakes (see Subsection 3.6.3.4).

The approach in considering piping system thermal flexibility is to route the pipe subjected to the thermal loads, other normal operating loads, seismic loads, and stress and fatigue limits. Determination of bounding leakage crack lengths for a LBB evaluation is made on the basis of a range of normal operating (NO) loads that span the loads for each pipeline evaluated.

3.6.3.2.3.2 Leakage Detection Systems (LDS)

The various means of leak detection discussed in Subsection 3.6.3.5.2.1 support the requirements of the LBB evaluation even though they may not be designed specifically for LBB. Regulatory Guide 1.45 (Reference 20) requires a LDS capable of detecting a 3.785 L/min (1.0 gpm) rate or less, independent of LBB requirements, and SRP 3.6.3 recommends a safety margin of ten (10) on the LDS capability. The LBB evaluation, however, depends on these “diverse measurement means,” their diverse sensitivities and accuracies, which constitute the LDS, in order to correlate a crack length to a flow rate 10 times the leak detection capability. See Subsection 3.6.3.3.2 for the detectable leak rate requirement for LBB evaluations.

3.6.3.2.3.3 Consideration of Potential for Degradation Sources

In order to meet the commitment of Subsection 3.6.3.4, the LBB evaluation considers pipe and weld material selection, significant thermal modes of operation, the environment in which the piping is routed, and potential for water hammer within the particular fluid system, as each relates to potential for degradation of the pipe.
Consideration of LBB is integrated into the process of selecting materials (for corrosion resistance), determining modes of operation (for reduction of loads from critical thermal transients), designing the piping system to preclude water hammer, and routing to minimize potential of failure of the pipe from indirect causes. Piping lines are evaluated for susceptibility to erosion, erosion/corrosion, erosion/cavitation, creep fatigue, cleavage type failure, and fatigue cracking. See Subsection 3.6.3.4 for discussions of consideration of degradation sources.

3.6.3.2.3.4 Consideration of Loading Conditions

Loads due to NO (dead weight, pressure, and normal steady state thermal conditions including, where applicable, long-term thermal stratification effects) are applied to the pipe section at each location selected for LBB evaluation to calculate a crack length that will result in 10 times the detectable leakage rate.

NO loads, critical thermal transients (including loads due to thermal stratification), SSE loads, and normal operation dynamic transient loads (such as from rapid valve closure) are considered in the stability analyses. SSE loads include the effects of SSE inertia plus seismic anchor motion (SAM), except that loads due to SAM are ignored when fractional SSE SAM loads are low in comparison to corresponding fractional SSE inertial loads.

A maximum design load is applied to the cracked pipe section in the stability analyses, along with the applicable load margin. Maximum design load is determined to be the largest of the following combined loads as discussed in Subsection 3.6.3.5.5.3;

a. Combined load of SSE loads, and normal operation dynamic transient loads such as from rapid valve closure with normal operating loads

b. Combined load of normal operating loads and stratified flow loads (where applicable)

3.6.3.3 Criteria for Leak-Before-Break

The LBB evaluation is consistent with the requirements set forth in SRP 3.6.3(Reference 4) and NUREG-1061, Volume 3(Reference 8)
3.6.3.3.1 Applicability of LBB

See Subsection 3.6.3.4 for discussions of LBB applicability with respect to consideration of degradation sources for the piping systems listed in Subsection 3.6.3.1.

3.6.3.3.2 Detectable Leakage Rate Requirement

As per Regulatory Guide 1.45, the detectable leakage rate requirement of the leak detection system is 3.785 L/min (1.0 gpm) or less. The leakage crack subjected to the crack stability analyses must leak at a rate ten times the capability of the LDS unless otherwise justified. Subsection 3.6.3.5.2.1 commits to these requirements of Regulatory Guide 1.45.

The LBB evaluations are based on a leak detection capability of 1.89 L/min (0.5 gpm), and a safety margin of 10.

3.6.3.3.3 Stability Analysis Acceptance Criteria

Stability analysis acceptance criteria are given in Subsection 3.6.3.6.

3.6.3.3.4 LBB Design Criteria Development

LBB acceptance criteria are developed for each of the selected piping systems using the procedure shown in Figure 3.6-2. LBB evaluations are performed to establish LBB acceptance criteria in terms of NO and maximum design loads for all locations determined in accordance with the selection criteria of Subsection 3.6.3.5.1. The LBB acceptance criteria are established in the form of an LBB piping evaluation diagram (PED), which is utilized to route, design and support the piping system.

The analyses done at the LBB design criteria development stage to create the LBB PED are performed using analytical methods of Subsection 3.6.3.5. The PED is further discussed in Subsection 3.6.3.5.5. The LBB PEDs are additional criteria to which the piping system is designed. Design of a piping system to the LBB requirements developed using the above approach assures that LBB has been demonstrated.
3.6.3.4 Potential Failure Mechanisms

3.6.3.4.1 Susceptibility to Failure from Erosion, Erosion/Corrosion, Erosion/Cavitation

Systems susceptible to erosion/corrosion pipe wall thinning are those with wet steam, flashing liquids, or liquid flow with high localized velocities. These factors are considered along with water chemistry and usage time to determine susceptibility and appropriate preventative methods.

3.6.3.4.1.1 Erosion/Corrosion Minimization

For systems susceptible to erosion/corrosion, the following methods are used to minimize degradation:

a. Additional wall thickness is specified to accommodate a limited amount of wall thinning without violating code requirements, if required.

b. The bulk fluid velocity is limited to prevent excessive erosion of the pipe wall.

c. Velocity guidelines may be increased on a case-by-case basis through the use of engineering evaluations that address the erosion/corrosion aspects and piping material selected in the design.

3.6.3.4.1.2 Leak-Before-Break Applicability to Piping

Use of high-quality steels, stainless steel, or stainless steel liners in the RCL, SL, DVI, and SC piping prevents erosion, erosion/corrosion, and erosion/cavitation. Additionally, water chemistry for the RCS is closely controlled and monitored. There is no evidence of unusual wall thinning in these pipes due to erosion, erosion/corrosion, or erosion/cavitation in pressurized water reactor (PWR) plants. Therefore, these pipes have a very low level of susceptibility to failure from these failure mechanisms.

3.6.3.4.2 Susceptibility to Failure from Water Hammer

NUREG-0927 (Reference 23) provides recommendations to be included in operating and maintenance procedures which include: A) prevention of rapid valve motion, B) proper filling and venting of water-filled lines and components, C) introduction of voids into waterfilled lines and components, D) introduction of steam or heated water that can flash

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into water filled lines, E) introduction of water into steam-filled lines or components, F) proper warm-up of steam-filled lines, G) proper drainage of steam-filled lines, and H) the effects of valve alignments on line conditions.

The implementations within the APR1400 design for each of the items in NUREG-0927 are described in Subsection 3.6.3.4.2.1 through Subsection 3.6.3.4.2.3. Each of items E) though H) in NUREG-0927 is not a cause to the water hammer in piping systems designed for LBB because there is no steam filled lines and valve alignments are not required. The COL applicant is to provide the procedure for initial filling and venting to avoid the known causes for water hammer in each piping system designed for LBB (COL 3.6(4)).

3.6.3.4.2.1 Reactor Coolant Loop (RCL) and Surge Line (SL)

There is a low potential for water hammer in the subcooled water solid portions of the RCS because these portions of the RCS are designed to preclude void formation. Safety valve discharge loads associated with the PZR have been identified and included in the component design basis. The reactor coolant gas vent system (RCGVS) is used to vent/discharge non-condensable gases and steam from the high points of the RCS. Non-condensable gases from the reactor vessel closure head and the pressurizer steam space will be vented during plant startup process to fill the RCS. Therefore, the RCL and SL piping have a low level of susceptibility to failure from water hammer.

3.6.3.4.2.2 Direct Vessel Injection (DVI) Line

The most likely cause for water hammer events in the DVI line is leaking check valves, allowing hot water to enter a low-pressure region and then flash into steam bubbles. The steam pocket thus formed would permit a steam pocket collapse type of water hammer to occur if it were suddenly pressurized by the addition of water to the low-pressure piping.

NUREG/CR-2781 (Reference 16) identifies four water hammer events from the NUREG/CR-2059 (Reference 17) database involving the safety injection system (SIS). EPRI research on water hammer events included these four events and two more related to the SIS as reported in EPRI NP-6766 (Reference 18). Five of these six events occurred in piping upstream of the injection check valves due to steam pocket collapse (3 events), filling of a voided line (1 event), and an unknown cause (1 event). The sixth event occurred in the low-head safety injection suction pump piping due to an unknown cause.
High-point vents provide for the proper venting of lines and pumps. If the piping is then pressurized (above the calculated leakage-induced temperatures/saturation pressures), the pressure coupled with the generally low temperature of the DVI system provides reasonable assurance that the lines will remain full and that steam bubbles will not develop near the check valves.

Further protection against this type of water hammer is provided administratively by monitoring the pressure in the injection line and flushing upon high pressure. Pressure indication and alarms are provided to alert the operator of an increase in pressure to 6.89 MPa (1,000 psig) (from a normal of approximately 4.27 MPa [620 psig]). Increased pressure is an indication of high-temperature RCS leakage past the DVI check valve. Upon an alarm, the operator opens the injection line drain valve to depressurize the injection line to the safety injection tank (SIT) pressure while replenishing the volume with subcooled water at containment ambient temperature. The replenishment is performed slowly so as not to exceed the makeup capability, therefore minimizing the potential for collapse of any steam pockets that may have formed.

Normal valve operation, pump startup, and pump trip create negligible fluid transient loads for the DVI system. The sixth event, as reported in EPRI NP-6766, is not a concern for the APR1400 because the SIS does not have a low-head safety injection pump. All results of all design basis events are mitigated with the use of four high-pressure pumps.

Based on system operating procedures that require venting of DVI lines and the low number and low severity of events reported for safety injection type systems in PWRs, the susceptibility to water-hammer-induced failures in the DVI system is low. Thus, the DVI system meets the screening criterion for the water hammer.

3.6.3.4.2.3 Shutdown Cooling Line

The PWR residual heat removal (RHR) system piping is susceptible only to a small number of the generic causes of water hammer such as rapid valve opening or closing and steam bubble collapse.

There is little potential for water hammer loading due to rapid valve opening or closing because there are no fast-acting valves in the shutdown cooling system (SCS), and a steam bubble is unlikely to form in the line. Under normal power operation, the valves in the line are closed and the fluid in the line is at ambient temperature. Thus, a low vapor pressure and steam bubble formation do not occur. During SC operation, the system is
open to the RCS and has the same vapor pressure as the RCS, which would be subcooled due to the hydrostatic head formed by the water and steam in the PZR. Therefore, steam bubble formation is precluded by the characteristics inherent to the system.

Even though there have been water hammer events reported in parts of the PWR RHR system, severe water hammer events are not expected for the SC line in the APR1400 design. Based on the low severity of the types of events to which the SCS is subject, the SCS meets the screening criterion for water hammer.

3.6.3.4.3 Susceptibility to Failure from Creep Fatigue

Creep fatigue is a concern for ferritic steel piping at operating temperatures above 370 °C (700 °F) and for austenitic stainless steel piping at operating temperatures above 425 °C (800 °F). Operating temperatures of the piping systems are below these limits and therefore not susceptible to creep fatigue failure.

3.6.3.4.4 Susceptibility to Failure from Corrosion

Materials used in the RCL (stainless steel cladding), SL, DVI, and SC piping are highly resistant to corrosion. Material selection, fabrication controls, and water chemistry controls provide reasonable assurance of resistance to corrosion.

To prevent intergranular stress corrosion attack on the austenitic stainless steel piping (SL, DVI, and SC), fabrication and operation controls are implemented. Primary water chemistry is controlled to minimize contaminants, and the dissolved oxygen is at a level that would normally preclude intergranular stress corrosion cracking and transgranular stress corrosion cracking.

Therefore, through material selection, chemistry control, and fabrication control, these pipes have a very low level of susceptibility to failure from corrosion.

According to EPRI MRP-111 (Reference 19), primary water stress corrosion cracking in alloy 52/152 butt welds in dissimilar-metal welds is unlikely. Welding procedures including repair procedure will be qualified by the COL applicant to minimize tensile stresses on the internal diameters and dilution effects. Weld repairs that will be in contact with the fluid will be made such that there will be compressive stress conditions on the wetted surface (COL 3.6(5)).
Refer to Subsections 5.2.3.2, 5.2.3.3, and 5.2.3.4 for water chemistry controls and fabrication of reactor coolant boundary components.

3.6.3.4.5 Susceptibility to Failure from Indirect Causes

Pipe degradation or failure from indirect causes such as fires, missiles, and component support failure is prevented by designing, fabricating, and inspecting to criteria that provide reasonable assurance of a low probability of the event or its impact on safety-related structures. The detailed description are provided below:

a. Seismic events: The LBB-applied piping systems are designed to Seismic Category I (see Section 3.2).

b. System over-pressurization: Over-pressure protection for RCS, primary side of auxiliary or emergency systems connected to the RCS, and secondary side of SG is described in Subsection 5.2.2.

c. Human error: Section 18.1 describes human factors engineering program to support the operator and minimize the potential for operator errors.

d. Fires: Fire prevention and protection are described in Subsection 9.5.1.

e. Flooding: Flood protection and evaluation are described in Section 3.4.

f. Missiles: Missile protection is described in Section 3.5.

g. Damages from moving equipment: The containment polar crane is designed to maintain its integrity without dropping its load during an SSE. Subsection 9.1.5 describes the overhead heavy load handling system.

h. Failures of structures, systems or components (SSC): The SSCs in close proximity to the LBB-applied piping system are safety-related and seismically designed (see Section 3.2).

3.6.3.4.6 Susceptibility to Cleavage-Type Failure

Cleavage-type failures are generally not a concern for the system operating temperatures and materials used for the RCL, SL, DVI, and SC piping. Reasonable assurance of the resistance to brittle cleavage-type failure is provided by fracture toughness test or ASME
Section III, Appendix G analysis, and maintained on the full scope of system operation based on the pressure-temperature limit curve. In addition, material tests (ASME Section III required toughness tests and J-Resistance [J-R] tests) show the materials for these pipelines to be highly ductile and highly resistant to cleavage-type failures at operating temperature.

3.6.3.4.7 Susceptibility to Failure from Fatigue Cracking

The RCL, SL, DVI, and SC piping is designed to meet the ASME Section III, NB fatigue criteria. All design basis transients identified in Subsection 3.9.1 are included in the detailed stress analyses. In the detailed stress analysis, the effects on thermal stratification and turbulence penetration are considered. If the impacts by these phenomena are sufficiently low, these phenomena are not considered in the design.

The potential for vibration-induced fatigue cracking within the RCL is primarily due to vibrations by the reactor coolant pumps (RCPs) operation. The RCP-induced vibrations are minimized by limiting pump shaft and frame vibrations during hot functional testing and operation. Also, piping vibrations are tested during initial test program as addressed in Subsection 3.9.2.1. During operation, RCP vibration monitoring system monitors the pump shaft and frame vibrations and provides the alarms to the operators in the main control room (see Subsection 7.7.1.5).

DVI and SC piping systems where LBB is applied are not operated during full power operation mode. Therefore, these system piping will not experience of vibration and result in vibration fatigue problem.

Therefore, these pipes have a very low susceptibility to failure from fatigue cracking.

3.6.3.4.8 Susceptibility to Failure from Thermal Aging

The LBB-applied piping systems for the APR1400 have no cast materials. The materials for the RCL are forged carbon steel and carbon weld metals, and they are known to have non-relevance to thermal aging. The austenitic stainless steel piping for SL, DVI, and SC are all forged products and have a low susceptibility to thermal aging. The welds in austenitic stainless steel piping are fabricated using the gas tungsten arc welding (GTAW) process, and weld metals are controlled to have low delta-ferrites, as described in Subsection 5.2.3.4.5.
Therefore, thermal aging of the LBB-applied piping systems for the APR1400 is not a concern.

3.6.3.4.9 Failure Prevention and Detection

3.6.3.4.9.1 Snubber Reliability

The functional requirements and design specifications provided to the supplier of snubbers contain the information described in Subsection 3.9.3. Snubber application and locations are determined during the design and tested as described in Subsection 3.9.6.

3.6.3.4.9.2 Inspections

For the piping systems identified in Subsection 3.6.3.1 to which LBB is applied, 100% of all welds will undergo a preservice inspection. Additionally, in accordance with the ASME Sections III and XI requirements for ASME Class 1 and 2 piping, both preservice and inservice inspection requirements are implemented to provide assurance of the integrity of those systems to which the LBB methodology is applied.

In accordance with ASME Section XI, Division 1, IWA-1500, and the requirements of 10 CFR 50.55a(g)(3)(i), biological shielding around the reactor coolant piping is designed to afford access to the circumferential and longitudinal welds for examination, as well as the transition piece-to-nozzle welds. The piping systems to which LBB is applied are provided with removable insulation in the areas of all welds and adjacent base metal requiring examination. The volumetric examinations are performed using ultrasonic techniques.

More information on ISI requirements is provided in Subsection 5.2.4.

3.6.3.5 Analysis

Guidance for the approach to LBB analysis is provided in SRP 3.6.3 (Reference 4) and NUREG-1061, Volume 3 (Reference 8). In these documents the use of ductile fracture mechanics is recommended. The J-integral and the tearing modulus are used to assess the stability of the postulated through-wall cracks. LBB PEDs are developed for the piping systems identified in Subsection 3.6.3.1. These diagrams constitute the crack stability acceptance criteria.
3.6.3.5.1 Leakage Crack Location

It is a regulatory requirement that LBB be applied to an entire piping system or analyzable portion thereof, typically segments located between anchor points. A survey of the piping is performed to determine the locations that have the least favorable combination of stress and material properties for base metal, weldments, nozzles, and safe ends. Locations of the highest maximum design stress are determined for each type of material present within the piping run in order to define the locations where the LBB evaluation is performed. Locations of the highest maximum design stress are based on the calculation of the combined stresses from the design basis load discussed in Subsection 3.6.3.2.3.4.

3.6.3.5.2 Leakage Crack Length

There are two major aspects to leak rate based on crack detection in addition to the crack opening size: leak detection capability and flow rate correlation for leakage through a crack.

3.6.3.5.2.1 Leak Detection System

NRC RG 1.45 (Reference 20) recommends a leak detection system that is capable of detecting a leakage rate of 3.785 L/min (1.0 gpm) or less to the reactor containment. The leak detection capability for the APR1400 is 1.89 L/min (0.5 gpm). Diverse measurement means are provided for leakage detection, including RCS inventory monitoring, sump level and flow monitoring, and measurement of airborne radioactive particulates and gases (refer to Subsection 5.2.5). The RCS primary water inventory balance method is used to detect leakage rates of 1.89 L/min (0.5 gpm) or less.

3.6.3.5.2.2 Flow Rate Correlation

The other major aspect of crack detection based on the leak rate, namely the flow rate correlation for leakage through a given crack size, cannot be predicted precisely. Variables such as surface roughness of the side walls of the crack, the nonparallel relationship of the side walls due to the elongated crack shape, and possible zigzag tearing of the material during crack formation introduce uncertainties in defining an exact flow rate correlation.

The leakage rate required to be detectable is 1.89 L/min (0.5 gpm) or less. The licensing guidelines in SRP 3.6.3 recommend a factor of 10 on the leakage rate for conservatism.
unless otherwise justified. The LBB evaluations of those piping systems listed in Subsection 3.6.3.1 are based on a leak of 18.9 L/min (5 gpm) with a safety margin of 10.

3.6.3.5.2.3  Leakage Crack Length Determination

It is necessary that hypothesized through-wall cracks open significantly to allow detection by normal leakage monitoring under normal full power loadings.

The leakage crack length for a required 18.9 L/min (5 gpm) flow depends upon the pipe loading, thermodynamic conditions, and assumed crack surface roughness conditions. The elastic-plastic estimation method of Reference 21 is used to find the crack opening displacement for a given loading. The PICEP program (Reference 22) is used to calculate the flow for a given crack length and loading. Crack morphology parameters used for PICEP program are shown in Table 3.6-7. For the purpose of generating analysis data for PEDs, a plot of moment versus crack length for an 18.9 L/min (5 gpm) flow is made using PICEP. This is done for each of the pipelines being evaluated for LBB. Each curve provides the relationship between normal operating loads (i.e., deadweight, thermal expansion, and pressure) and the crack length that gives a 18.9 L/min (5 gpm) flow.

At the interface between nozzle and surge line, consideration of the nozzle requires an iterative procedure to find an appropriate crack length which leaks at 18.9 L/min (5 gpm) and employs both the finite element model used for the crack stability analysis and the PICEP program. Since the stiffness of the nozzle is included in the stability analysis, it is also included in the leakage calculation. The iterative procedure used for calculating crack length at the nozzle/pipe interface is as follows:

a. Assume a crack length in finite element model

b. Apply normal operating loads to finite element model and calculate the crack opening area

c. Using PICEP program with the same crack length, vary the applied moment until the crack opening area becomes the same area as calculated with the finite element model.

d. If the calculated flow by PICEP program is greater than 18.9 L/min (5 gpm), the crack length is decreased and go to step ‘a’. If the calculated flow is less than 18.9 L/min, keep the crack length.
L/min (5 gpm), the crack length is increased and go to step ‘a’. If the flow is 18.9 L/min (5 gpm), stop.

The moment vs. crack length curves for the RCL, SL, DVI and SC lines listed in Subsection 3.6.3.1 are shown in Figures 3.6-15 through 3.6-22. These curves are the crack lengths associated with 18.9 L/min (5 gpm) flow for a given moment. An analysis using a longer crack at a given moment results in more than 18.9 L/min (5 gpm) flow.

3.6.3.5.3 Material Properties

3.6.3.5.3.1 Stress-Strain Curves

The hot leg (HL) and cold leg (CL) are typically fabricated from SA-516 Gr. 70 or SA-508 Gr. 1a. The stress-strain curve is taken from the LBB test data for the reference plant. The Ramberg-Osgood material characterization described in Table 3.6-2 is for the detectable crack length calculation and the Ramberg-Osgood fit to these data is described in Figure 3.6-3. The lower-bound stress-strain curve described in Figure 3.6-7 is applied for the crack stability evaluation.

The SL is typically fabricated with SA-312 Type 347 or TP316N stainless steel. The stress-strain curves are taken from the LBB test data for the reference plant bounds for the stainless steel used in the design. The stress-strain data are shown in Table 3.6-2 for the detectable crack length calculation. Figure 3.6-4 shows the Ramberg-Osgood fit to the data. In the crack stability evaluation, the lower-bound data are applied and the stress-strain curve is described in Figure 3.6-8.

The DVI and SC lines are typically fabricated with SA-312 TP304 and SA-312 TP316 stainless steel. The stress-strain curves are taken from the LBB test data for the reference plant. In the calculation of the detectable crack length, the bound data for each material are applied as described in Table 3.6-3. The Ramberg-Osgood data fit is shown in Figures 3.6-5 and 3.6-6. In the determination of the detectable crack length, the base metal test data for each material specification are applied. Otherwise, the lower-bound data are chosen from base and weld metal per material specification. The data are described in Figures 3.6-9 and 3.6-10.
3.6.3.5.3.2 Material J-Resistance Curves

The material J-R curves are taken from the LBB test data for the reference plant for the HL, CL, SL, DVI, and SC lines. The J-R material curve used for the HL and CL is shown in Figure 3.6-11. A fit to the data used in the stability evaluation is shown in the figure. This J-R curve bounds the material toughness behavior in any of these pipelines of the reference plant.

In order to provide reasonable assurance that LBB is satisfied for SL, DVI and SC lines, which are relatively small-diameter pipes, GTAW is specified for all shop and field welds. The material J-R curves are plotted in Figure 3.6-12 for the SL, Figure 3.6-13 for TP304 stainless steel of the DVI line, and Figure 3.6-14 for TP316 stainless steel of the DVI and SC lines.

The parameters on J-R curves for LBB application piping are shown in Table 3.6-6. However, the lower-bound data chosen from LBB test data may be used for conservatism.

3.6.3.5.4 Finite Element Model Description

3.6.3.5.4.1 Geometry and Boundary Conditions

The finite element model for a typical crack stability evaluation is shown in Figure 3.6-23. A close-up of the crack tip area is shown in Figure 3.6-24. The finite element model is simply a means for applying the pressure and moment loading to a section of pipe containing the hypothetical crack at some location in the pipeline. Two planes of symmetry are used to minimize the size of FE models for RCL, DVI, and SC pipes and intermediate pipe of SL. Therefore, each model represents one quarter of the pipe as shown in Figure 3.6-23(b). One plane of symmetry for the crack near the hot leg surge nozzle and pressurizer surge nozzle as shown in Figure 3.6-23(a) is used to minimize the model size, meaning that one half of the nozzle and pipe is modeled.

The length of the pipe is chosen to be at least five pipe diameters in order that the point of load application is not close to the crack tip region. The mesh uses 20-node isoparametric brick reduced integration elements. Boundary conditions are imposed on the model based upon symmetry and crack location. The crack surface area is free from constraint.
3.6.3.5.4.2 Loadings

The internal pressure appropriate to the normal operating conditions of each piping system is applied to the inner surface of the pipe, and the one half of internal pressure is applied to the crack face to account for the pressure drop across the crack. An axial end load traction, which when integrated over the pipe cross-sectional area, is equal to the continuity axial force, is applied to the far end of the pipe. Moments are applied as a linearly varying traction to the far end of the pipe.

3.6.3.5.4.3 J-Integral

The stability of through-wall cracks is evaluated using the J-integral technique. The J-integral parameter is related to the energy release rate at the crack tip. For APR1400 piping system listed in Subsection 3.6.3.1, the J-integral is determined by finite element analysis for maximum design load included in the crack stability evaluation using ABAQUS program (See Subsection 3.9.1.2.1.1).

3.6.3.5.4.4 Stability Evaluation

The stability of the cracked pipes is assessed by comparing the J-integral value due to the applied loads on the pipe to the material crack resistance. The stability criterion for ductile crack extension that is used is as follows:

\[
J_{\text{applied}} < J_{\text{material}}
\]

and

\[
\left( \frac{dJ}{da} \right)_{\text{applied}} < \left( \frac{dJ}{da} \right)_{\text{material}}
\]

Then, reasonable assurance of crack stability is provided.

The change in J-integral with crack length “a” is determined by analyzing several crack lengths in the region of interest. For a leakage crack of length “a,” crack lengths, “a,” “a-d,” and “a+d,” are analyzed as shown in Figure 3.6-25. Similarly, the change in J-integral with crack length in the region of length “2a” is determined by analyzing cracks with lengths ”2a,” “2a-d,” and “2a+d.” This method provides the derivative information in the two regions of interest. The variation of J with crack length in the region of “a” and
“2a” is plotted along with the material curve. Evaluation of the plots allows for direct verification of the stability criteria.

The evaluations are performed for the locations chosen to envelop all limiting cases. The pipes with the leakage crack length subject to loads of $\sqrt{2}$ times the maximum design load and the pipes with crack length twice the leakage crack length with loads of the maximum design load are demonstrated to have significant margin between the material curve and the loading curve, indicating that all pipe locations satisfy the LBB crack stability criteria.

An acceptable alternative method for the margin on loads and margin on crack length evaluations is to combine each component of the absolute values for the deadweight, thermal expansion, pressure, SSE (inertial), and seismic anchor motion (SAM) loads. This method is referred to as the absolute summation of loads method. If this alternative method is used, the margin on load for the leakage crack size is reduced from $\sqrt{2}$ to 1. The margin on crack length (2 times the leakage crack size) remains the same.

For each load step in the analysis, the loading curve as a function of crack length is fit to a quadratic:

$$J(a) = C_1 a^2 + C_2 a + C_3$$

The values at “a” and “a ± d” provide the boundary conditions necessary to evaluate the constants $C_1$, $C_2$, and $C_3$. At each loading point, the function is differentiated. This provides the $dJ/da$ values for the loading curve. The material curves $J(a)$, $dJ(a)/da$ are evaluated at increasing crack extension. The loading functions $J(a)$, $dJ(a)/da$ are evaluated at either “a” or “2a,” whichever crack length is being evaluated. Each point on the $J$ versus $dJ/da$ loading curve corresponds to a different load state. As long as the loading curve stays below the material curve,

$$J_{\text{load}} < J_{\text{material}}$$

and

$$\frac{dJ_{\text{load}}}{da} < \frac{dJ_{\text{material}}}{da}$$
the crack growth is stable.  For the case of increasing load, the loading curve eventually intersects the material curve.  At this point, the crack would experience unstable crack growth.  At this point of instability,

\[ J_{\text{load}} = J_{\text{material}} \]

and

\[ \frac{dJ_{\text{load}}}{da} = \frac{dJ_{\text{material}}}{da} \]

Development of the \( J \) versus \( dJ/da \) diagram for determining points of instability is shown in Figure 3.6-26.

3.6.3.5.5  Piping Evaluation Diagram

3.6.3.5.5.1  Inputs for Development of Piping Evaluation Diagram

The following data forming the basis for developing LBB PEDs are summarized in Table 3.6-4 and Table 3.6-5:

a. Pipe size(diameter and thickness)

b. Materials

c. Weld type

d. Material properties

e. Normal operating temperature

f. Normal operating pressure

g. Leak detection capability

3.6.3.5.5.2  Constructing a Leak-Before-Break Piping Evaluation Diagram

The method by which LBB PEDs are constructed allows for the evaluation of the piping system in advance of the final piping analysis, incorporating LBB considerations into the piping design.  The LBB PED for each pipe size, material, and pressure is prepared prior
to the piping design and analysis and is used to evaluate critical points in the pipeline. The PED is constructed to allow the maximum design load to be plotted vs. the normal operating (NO) load.

The maximum design load at any time during the plant operation is the loading used in the stability analysis. Traditionally, this loading has been NO+SSE. However, the combination of the NO load and the largest of the design loads are used in the stability analysis. In the case of the SL, for example, the line is evaluated for the larger of either NO+SSE or stratified flow (SF). Subsection 3.6.3.5.5.3 describes the load combination method to determine NO and maximum design loads.

The LBB piping evaluation diagram requires performing two complete LBB evaluations. The evaluations are for two NO loads that span the typical loadings for the line under consideration. A completed typical diagram is shown in Figure 3.6-27. The procedure used for generating that figure is as follows:

a. Choose NO1 load

b. Determine leakage crack length (a_l and 2a_l) with corresponding NO1 load by the approach provided in Subsection 3.6.3.5.2.3.

c. Determine the critical moment, M_{critical} for a_l and 2a_l by the approach provided in Subsection 3.6.3.5.4.

d. Calculate the maximum allowable load from the critical moment, M_{critical} for a_l and 2a_l

$$M_{critical} = \sqrt{2} M_{allowable1} \quad (a_l \text{ analysis})$$

$$M_{allowable1} = \frac{M_{critical}}{\sqrt{2}} \quad \text{and}$$

$$M_{critical} = M_{allowable1} \quad (2a_l \text{ analysis})$$

$$M_{allowable1} = M_{critical}$$

e. Plot M_{allowable1} values at NO1 load for a_l and 2a_l analyses, respectively. This corresponds to the points labeled “1” in Figure 3.6-27.

f. Repeat steps a to e for NO2 load. The results are shown in Figure 3.6-27, labeled “2”.
If the absolute summation of loads method is used to evaluate the margin on load and margin on crack length, the PED for the a1 analysis is constructed using the formulas $M_{critical} = M_{allowable1}$ for the points labeled “1” and $M_{critical} = M_{allowable2}$ for the points labeled “2”.

3.6.3.5.5.3 Method of Load Combination

Normal operating load and maximum design load are applied to the LBB evaluation using LBB PED.

Normal operating load is combined algebraically as follows:

a. Piping systems except for surge line:

$$(M_i)_{NO} = (M_i)_{Deadweight} + (M_i)_{Thermal-100\%} + (M_i)_{Pressure-100\%}$$

b. Surge Line:

$$(M_i)_{NO} = (M_i)_{Deadweight} + (M_i)_{Thermal-100\%} + (M_i)_{Pressure-100\%} + (M_i)_{stratified-100\%}$$

where $M_i$ is the moment of i-component ($i=1,2,3$), and the subscript, Thermal-100% means 100% power linear thermal expansion, Pressure-100% means the pressure load at full-power operation, and Stratified-100% means the thermal stratified load at full-power operation.

Maximum design load is combined as follows:

a. Piping systems except for surge line

$$(M_i)_{combined} = \left| (M_i)_{Deadweight} + (M_i)_{Thermal-100\%} + (M_i)_{Pressure-100\%} \right| + \left| (M_i)_{SSE} \right|$$

Where SSE load includes both SSE inertia load and SSE seismic anchor motion(SAM) load. The subscript, Thermal-100% means 100% power linear thermal expansion.

b. Surge Line:

The larger of the following two loads is considered.
\[(M_i)_{combined} = \left[(M_i)_{Deadweight} + (M_i)_{Thermal-100\%} + (M_i)_{Pressure-100\%} + (M_i)_{Stratified-100\%}\right] + |(M_i)_{SSE}|\]

Where SSE load includes both SSE inertia load and SSE seismic anchor motion (SAM) load. The subscript, Thermal-100\% means 100\% power linear thermal expansion, and Stratified-100\% means the thermal stratified load at full-power operation.

\[(M_i)_{combined} = (M_i)_{Deadweight} + (M_i)_{Thermal-hp} + (M_i)_{Pressure-100\%} + (M_i)_{Max. stratified}\]

Where \(M_i\) is the moment of \(i\)-component \((i=1,2,3)\), and subscript, Thermal-hp means linear thermal expansion at plant heatup, and Max. stratified means maximum thermal stratification load.

To determine NO and maximum design moments for LBB PED input, only the transverse bending moments are used.

\[\left[(M_i)_{combined} = \sqrt{(M_2)^2_{combined} + (M_3)^2_{combined}}\right]\]

### 3.6.3.5.5.4 Using a Leak-Before-Break Piping Evaluation Diagram

Once the lines marking the acceptable areas of allowable piping loads are plotted as described in Subsection 3.6.3.5.5.2, normal operating loads and corresponding maximum design loads for the critical piping locations are calculated as provided in Subsection 3.6.3.5.5.3, and plotted on the piping evaluation diagram. The critical locations are selected as described in Subsection 3.6.3.5.1.

Figure 3.6-28 shows how the plot is used for a hypothetical line. In this example, three points failed LBB and one point passed LBB. The reasons for each failure are given in the figure. The piping design can then be revised using the results (e.g., lowering the SSE response load by rerouting or by adding a snubber). Further review may result in other options for reducing the loads.

### 3.6.3.6 Results

LBB PEDs for piping systems listed in Subsection 3.6.3.1 provide LBB acceptance criteria for these piping systems and are shown in the figures as listed below:
These criteria are based on piping design parameters given in Table 3.6-4 and Table 3.6-5. Analyses of preliminary design of these piping systems have demonstrated that the LBB criteria are met. These criteria require the following:

a. Leakage rate from the postulated leakage crack is detectable by the leakage detection system in the containment. The amount of leakage is detectable with a safety margin of at least a factor of 10 unless otherwise justified.

b. Cracks of the length that leak at the rate given above can withstand the maximum design loads with a safety factor of at least $\sqrt{2}$. Alternatively, cracks of the length that leak at the rate given above can withstand the absolute combination of the maximum design loads with a factor of one.

c. Cracks twice as long as those addressed above remain stable when subjected to the maximum design loads.

Site-specific information will demonstrate that the final detailed design parameters of each piping system are consistent with those given in Tables 3.6-4 and 3.6-5, and the final detailed design meets the LBB criteria of Figures 3.6-29 through 3.6-36. If design parameters for a piping system are not enveloped by the parameters in Tables 3.6-4 and 3.6-5, a new PED for that piping system will be constructed using the methodology given in
this subsection, and the piping design will be revised, as necessary, to meet the LBB criteria of the new PED. If a PED given in Figures 3.6-29 through 3.6-36 is applicable to the detailed design of a piping system, but the detailed design does not meet the LBB criteria of the PED, the design will be revised to meet the LBB criteria of the PED.

Reconciliation of the as-built piping systems with the final design is documented in an LBB evaluation report. The LBB evaluation report contains the results of the LBB evaluations for as-built piping. The LBB evaluations use the methods described in Subsection 3.6.3. Each as-built piping system qualified for LBB is reconciled by demonstrating that:

a. The as-built piping system meets the screening criteria of Subsection 3.6.3.4.

b. The dimensional and material properties of the as-built piping system are consistent with the parameters used in the development of the final LBB PED(s) for that piping system.

c. The as-built piping responses meet the ASME Code allowable and the final LBB PED criteria.

If the absolute summation of loads method is used, the PEDs will be reconstructed, and the piping design, evaluation, and reconciliation will be based on the reconstructed PEDs.

3.6.4 Combined License Information

COL 3.6(1) The COL applicant is to identify the site-specific SSCs that are safety-related or required for safe shutdown that are located near high- and moderate-energy piping systems and that are susceptible to the consequences of piping failures.

COL 3.6(2) The COL applicant is to provide a list of site-specific high- and moderate-energy piping systems including layout drawings and protection features and the failure modes and effects analysis for safe shutdown due to the postulated HELBs.

COL 3.6(3) The COL applicant is to confirm that the bases for the LBB acceptance criteria are satisfied by the final as-built design and materials of the piping systems as site-specific evaluations, and is to provide the information including LBB evaluation report for the verification of LBB analyses.
COL 3.6(4) The COL applicant is to provide the procedure for initial filling and venting to avoid the known causes for water hammer in each piping system designed for LBB.

COL 3.6(5) The COL applicant is to provide the information on welding of Alloy 52/52M/152 concerning the residual stress and dilution effects of welds.

3.6.5 References


# Table 3.6-1

## High- and Moderate-Energy Fluid Systems

<table>
<thead>
<tr>
<th>System (1)(2)</th>
<th>High Energy</th>
<th>Moderate Energy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Coolant System</td>
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<tr>
<td>Reactor Coolant Gas Vent System</td>
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<tr>
<td>Safety Injection System</td>
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<td></td>
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<tr>
<td>Shutdown Cooling System</td>
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</tr>
<tr>
<td>Chemical and Volume Control System</td>
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<td>Steam Generator Blowdown System (3)</td>
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<tr>
<td>Component Cooling System</td>
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<td>Spent Fuel Pool Cooling and Cleanup System</td>
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<tr>
<td>Process and Effluent Radiation Monitoring and Sampling System</td>
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<td>Process and Post-Accident Sampling System</td>
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<td>Containment Spray System</td>
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<td>Essential Service Water System</td>
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<td>Condensate and Feedwater System</td>
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<td>Auxiliary Steam System</td>
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<td>Emergency Diesel Generator System (4)</td>
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<td>Fire Protection System</td>
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<td>Equipment and Floor Drainage System</td>
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<td>Essential Chilled Water System</td>
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<td>Plant Chilled Water System</td>
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<tr>
<td>Compressed Air System</td>
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</table>

(1) Systems classified as high-energy are either totally or partially high-energy. If portions of system are high-energy, it is classified as high-energy system. The portions of high-energy piping are identified in Figures 3.6-1 through 3.6-12.

(2) Systems or portions of systems outside the containment building and auxiliary building are excluded from this table.

(3) Wet layup recirculation system is classified as moderate-energy system.

(4) Subsystems other than an EDG engine starting air system are classified as moderate-energy systems.

(5) Subsystems other than an auxiliary feedwater pump turbine subsystem are classified as moderate-energy systems.
Table 3.6-2

Material Constants (Reactor Coolant Loop and Surge Line)

Security-Related Information – Withhold Under 10 CFR 2.390
Material Constants (Direct Vessel Injection and Shutdown Cooling Line)

Security-Related Information – Withhold Under 10 CFR 2.390
Table 3.6-4

Design Parameters Used in LBB Evaluations (Reactor Coolant Loop and Surge Line)

Security-Related Information – Withhold Under 10 CFR 2.390
APR1400 DCD TIER 2

Table 3.6-5

Design Parameters Used In LBB Evaluation (Direct Vessel Injection and Shutdown Cooling Line)

Security-Related Information – Withhold Under 10 CFR 2.390
Table 3.6-6

Parameters on J-R Curves

Security-Related Information – Withhold Under 10 CFR 2.390
APR1400 DCD TIER 2

Table 3.6-7

Crack Morphology Parameters

Security-Related Information – Withhold Under 10 CFR 2.390
### Table 3.6-8 (1 of 6)

**High-Energy Line and Break Location**

<table>
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<tr>
<th>Subsystem</th>
<th>Room Number</th>
<th>High-Energy Line</th>
<th>Break location&lt;sup&gt;1)&lt;/sup&gt;</th>
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<tr>
<td>9RC101</td>
<td>136C02</td>
<td>RCS cold leg (RC003AD30) – PZR spray nozzle</td>
<td>Terminal End Figure 3.6-37 (1 and 2 of 2)</td>
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<tr>
<td>9RC101</td>
<td>136C02</td>
<td>RCS cold leg (RC003AB30) – PZR spray nozzle</td>
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<tr>
<td>9RC101</td>
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<td>Charging side regenerative HX outlet – PZR spray line connection</td>
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<td>9RC101</td>
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<td>PZR – PZR spray line connection (V200/V201)</td>
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<td>9RC101</td>
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<td>PZR – PZR spray line connection (V202/V203)</td>
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<td>9RC103</td>
<td>100C02B</td>
<td>RCS cold leg (RC002AD30) – RCS drain line (RC008AD2)</td>
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<td>9RC104</td>
<td>100C02A</td>
<td>RCS cold leg (RC002AA30) – RCS drain line (RC008AA2)</td>
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<td>9RC105</td>
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<td>9RC105</td>
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<td>RCS drain line letdown line (RC010A2) – Regenerative HX letdown line inlet nozzle</td>
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<td>9RC106</td>
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<td>9RG101</td>
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<td>RCS gas vent system POSRV inlet vent line – Vent line isolation valve (V412/413)</td>
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<td>9SI101</td>
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<td>SI tank (1A) vent line nozzle (SI009AA12) – RCS DVI nozzle (RC005AA12)</td>
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<td>Terminal End Figure 3.6-39 (1 of 2) Figure 3.6-37 (1 of 2)</td>
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**Note:**
1) There are no intermediate break locations of the piping system taken to the graded approach as a result of stress analysis and cumulative usage factor analysis for ASME Section III, Class 1 piping.
### Table 3.6-8 (2 of 6)

<table>
<thead>
<tr>
<th>Subsystem</th>
<th>Room Number</th>
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<td>9CV106</td>
<td>100C02B</td>
<td>Regenerative HX letdown line outlet nozzle – Letdown HX letdown line inlet nozzle / isolation valve inside containment (V363)</td>
<td>Terminal End</td>
</tr>
<tr>
<td>9CV107</td>
<td>100C02B</td>
<td>Letdown HX letdown line outlet nozzle – Letdown line at containment penetration (PC402)</td>
<td>Terminal End</td>
</tr>
<tr>
<td>9CV622</td>
<td>114C01A</td>
<td>RCP seal water line at containment penetration (PC304) – 2&quot;x1” reducer</td>
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<tr>
<td>9SD101</td>
<td>100C02A 136C01A</td>
<td>SG1 wet layup nozzle – Wet layup isolation valve (V1119)</td>
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<tr>
<td>9SD102</td>
<td>100C02B 136C01B</td>
<td>SG2 wet layup nozzle – Wet layup isolation valve (V1120)</td>
<td>Terminal End</td>
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<td>9SD103</td>
<td>100C02A</td>
<td>SG1 blowdown line nozzle – Blowdown line at containment penetration (PC911)</td>
<td>Terminal End</td>
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<tr>
<td>9SD104</td>
<td>100C02B</td>
<td>SG2 blowdown line nozzle – Blowdown line at containment penetration (PC921)</td>
<td>Terminal End</td>
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### Table 3.6-8 (4 of 6)

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<tr>
<td>9SD215</td>
<td>137A31C</td>
<td>Blowdown line at containment penetration (PC911) – Blowdown flash tank compartment wall</td>
<td>Terminal End</td>
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<tr>
<td>9SD216</td>
<td>137A31D</td>
<td>Blowdown line at containment penetration (PC921) – Blowdown flash tank compartment wall</td>
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<tr>
<td>9MS101</td>
<td>136C01B 100C02B</td>
<td>SG2 steam vent nozzle – SG2 main steam line at containment penetration (PC621)</td>
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<tr>
<td>9MS269</td>
<td>137A31D</td>
<td>SG2 main steam line at containment penetration (PC621) – SG2 main steam valve house anchor wall penetration</td>
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<tr>
<td>9MS288</td>
<td>137A31D</td>
<td>Main steam line (9M269) condensate drain line nozzle – Main steam valve house anchor wall penetration</td>
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<tr>
<td>9MS102</td>
<td>156C01 136C01B</td>
<td>SG2 steam vent nozzle – SG2 main steam line at containment penetration (PC622)</td>
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<tr>
<td>9MS270</td>
<td>137A31D</td>
<td>SG2 main steam line at containment penetration (PC622) – SG2 main steam valve house anchor wall penetration</td>
<td>Terminal End</td>
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<tr>
<td>9MS287</td>
<td>137A31D</td>
<td>Main steam line (9M270) condensate drain line nozzle – Main steam valve house anchor wall penetration</td>
<td>Terminal End</td>
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<tr>
<td>9MS275</td>
<td>137A31D</td>
<td>SG2 aux-feedwater pump turbine steam line connection – pipe layout area boundary wall</td>
<td>Terminal End</td>
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<tr>
<td>9MS103</td>
<td>136C01A 100C02A</td>
<td>SG1 steam vent nozzle – SG1 main steam line at containment penetration (PC612)</td>
<td>Terminal End</td>
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<tr>
<td>9MS271</td>
<td>137A31C</td>
<td>SG1 main steam line at containment penetration (PC612) – SG1 main steam valve house anchor wall penetration</td>
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<td>9MS286</td>
<td>137A31C</td>
<td>Main steam line (9M271) condensate drain line nozzle – Main steam valve house anchor wall penetration</td>
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<tr>
<td>9MS273</td>
<td>137A31C</td>
<td>SG1 aux-feedwater pump turbine steam line connection – pipe layout area boundary wall</td>
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## Table 3.6-8 (5 of 6)

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<tr>
<td>9MS104</td>
<td>136C01A, 100C02A</td>
<td>SG1 steam vent nozzle – SG1 main steam line at containment penetration (PC611)</td>
<td>Terminal End</td>
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<tr>
<td>9MS272</td>
<td>137A31C</td>
<td>SG1 main steam line at containment penetration (PC611) – SG1 main steam valve house anchor wall penetration</td>
<td>Terminal End</td>
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<tr>
<td>9MS285</td>
<td>137A31C</td>
<td>Main steam line (9M272) condensate drain line nozzle – Main steam valve house anchor wall penetration</td>
<td>Terminal End</td>
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<tr>
<td>9FW209</td>
<td>137A31C</td>
<td>SG1 economizer main steam valve house anchor wall – Economizer line at containment penetration (PC511)</td>
<td>Terminal End</td>
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<tr>
<td>9FW101</td>
<td>136C01A</td>
<td>Economizer line at containment penetration (PC511) – SG1 economizer injection nozzle</td>
<td>Terminal End</td>
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<tr>
<td>9FW601</td>
<td>100C02A</td>
<td>SG1 secondary drain line (FW066AA2) nozzle – Drain line isolation valve (V2122)</td>
<td>Terminal End</td>
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<tr>
<td>9FW627</td>
<td>100C02A</td>
<td>SG1 secondary drain line (FW066AB2) nozzle – Drain line isolation valve (V2123)</td>
<td>Terminal End</td>
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<tr>
<td>9FW219</td>
<td>137A31C</td>
<td>SG1 downcomer main steam valve house anchor wall – SG1 downcomer at containment penetration (PC512)</td>
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<td>9AF101</td>
<td>136C01A</td>
<td>SG1 downcomer at containment penetration (PC512) / aux-feedwater check valve (V1008A/V1007A) – SG1 downcomer injection nozzle</td>
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<tr>
<td>9FW210</td>
<td>137A31D</td>
<td>SG2 economizer main steam valve house anchor wall – Economizer line at containment penetration (PC521)</td>
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<td>9FW102</td>
<td>136C01B</td>
<td>Economizer line at containment penetration (PC521) – SG2 economizer injection nozzle</td>
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<tr>
<td>9FW602</td>
<td>100C02B</td>
<td>SG2 secondary drain line (FW066AC2) nozzle – Drain line isolation valve (V2124)</td>
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<tr>
<td>9FW628</td>
<td>100C02B</td>
<td>SG2 secondary drain line (FW066AD2) nozzle – Drain line isolation valve (V2125)</td>
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Table 3.6-8 (6 of 6)

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<td>9FW220</td>
<td>137A31D</td>
<td>SG2 downcomer main steam valve house anchor wall – SG2 downcomer at containment pen. (PC522)</td>
<td>Terminal End</td>
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<td>Figure 3.6-43 (1 of 1)</td>
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<tr>
<td>9AF102</td>
<td>136C01B</td>
<td>SG2 downcomer at containment penetration (PC522) / aux-feedwater check valve (V1008B/V1007B) – SG2 downcomer injection nozzle</td>
<td>Terminal End</td>
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<td>Figure 3.6-43 (1 of 1)</td>
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## APR1400 DCD TIER 2

### Table 3.6-9 (1 of 8)

System-Specific High-Energy Line Break Protection

<table>
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<tr>
<th>Subsystem</th>
<th>Break Number</th>
<th>Break Type</th>
<th>Break Location</th>
<th>Protection of Essential Target</th>
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<tr>
<td>Reactor Coolant System (Not applied for Leak-Before-Break): Terminal End Break</td>
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<tr>
<td>9RC101</td>
<td>BT-RC-01</td>
<td>C</td>
<td>Figure 3.6-37 (2 of 2)</td>
<td>PW ²) No PW occurs because of geometric configuration (limited movement of the broken pipe end)</td>
</tr>
<tr>
<td></td>
<td>BT-RC-02</td>
<td>C</td>
<td>Figure 3.6-37 (2 of 2)</td>
<td>JI ³) No essential target</td>
</tr>
<tr>
<td></td>
<td>BT-RC-03</td>
<td></td>
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<tr>
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<td>BT-RC-04</td>
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<tr>
<td>9RC101</td>
<td>BT-RC-05</td>
<td>C</td>
<td>Figure 3.6-37 (2 of 2)</td>
<td>PW Pressurizer compartment wall (WCE11-02) (designed to withstand PW impact load)</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>JI Top of pressurizer (designed to withstand JI load)</td>
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<tr>
<td></td>
<td>BT-RC-06</td>
<td>C</td>
<td>Figure 3.6-37 (1 of 2)</td>
<td>PW No essential target</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>JI RCS discharge leg (designed to withstand JI load)</td>
</tr>
<tr>
<td></td>
<td>BT-RC-07</td>
<td>C</td>
<td>Figure 3.6-37 (1 of 2)</td>
<td>PW Steel structure (2CS101) (designed to withstand PW impact load)</td>
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<tr>
<td></td>
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<td></td>
<td>JI RCS discharge leg (RC003AB30) (designed to withstand JI load)</td>
</tr>
<tr>
<td>9RC104</td>
<td>BT-RC-08</td>
<td>C</td>
<td>Figure 3.6-37 (1 of 2)</td>
<td>PW RCP 01A concrete pedestal, El. 100’ floor (SCE41-01) (designed to withstand PW impact load)</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>JI RCS suction leg (RC002AA30) (designed to withstand JI load)</td>
</tr>
<tr>
<td>9RC106</td>
<td>BT-RC-09</td>
<td>C</td>
<td>Figure 3.6-37 (1 of 2)</td>
<td>PW RCP 02B concrete pedestal, El. 100’ floor (SCE11-01) (designed to withstand PW impact load)</td>
</tr>
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<td>JI RCS suction leg (RC002AB30) (designed to withstand JI load)</td>
</tr>
<tr>
<td>9RC105</td>
<td>BT-RC-10</td>
<td>C</td>
<td>Figure 3.6-37 (1 of 2)</td>
<td>PW RCP 01B concrete pedestal, El. 100’ floor (SCE31-01) (designed to withstand PW impact load)</td>
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<td>JI RCS suction leg (RC002AC30) (designed to withstand JI load)</td>
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Notes:

1) C and L stand for circumferential and longitudinal breaks, respectively.
2) Pipe whip
3) Jet impingement
### Table 3.6-9 (2 of 8)

<table>
<thead>
<tr>
<th>Subsystem</th>
<th>Break Number</th>
<th>Break Type</th>
<th>Break Location</th>
<th>Protection of Essential Target</th>
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<tbody>
<tr>
<td><strong>Reactor Coolant System (Not applied for Leak-Before-Break): Terminal End Break</strong></td>
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<td>9RC103</td>
<td>BT-RC-11</td>
<td>C</td>
<td>Figure 3.6-37  (1 of 2)</td>
<td>PW RCP 02B concrete pedestal, El. 100’ floor (SCE21-01) (designed to withstand PW impact load)</td>
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<td>JI RCS suction leg (RC002AD30) (designed to withstand JI load)</td>
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<td>BT-RC-12</td>
<td>C</td>
<td>Figure 3.6-37  (1 of 2)</td>
<td>PW Steel structure (3CS421) (designed to withstand PW impact load)</td>
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<tr>
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<td>JI Shutdown cooling line (RC004AA16) (designed to withstand JI load)</td>
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<td>BT-RC-13</td>
<td>C</td>
<td>Figure 3.6-37  (1 of 2)</td>
<td>PW Steel structure (3CS221) (designed to withstand PW impact load)</td>
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<td>JI Shutdown cooling line (RC004AB16) (designed to withstand JI load)</td>
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<td>BT-RC-14</td>
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<td>Figure 3.6-37  (1 of 2)</td>
<td>PW El. 100’ floor (SCE21-01) (designed to withstand PW impact load)</td>
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<td>JI Shutdown cooling line (RC004AB16) (designed to withstand JI load)</td>
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<td>BT-RC-15</td>
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<td>Figure 3.6-37  (1 of 2)</td>
<td>PW Hot leg injection line (SI010EC4) (designed to withstand PW impact load)</td>
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<td>JI RCP cold leg (RC003AA30) (designed to withstand JI load)</td>
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<tr>
<td><strong>Safety Injection/Shutdown Cooling System (Not applied for Leak-Before-Break): Terminal End Break</strong></td>
<td>9SI619</td>
<td>C</td>
<td>Figure 3.6-39  (2 of 2)</td>
<td>PW No PW occurs.(not enough reserved energy to move the broken pipe end)</td>
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<td>9SI620</td>
<td>BT-SI-02</td>
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<td>Figure 3.6-39  (1 of 2)</td>
<td>PW No PW occurs.(not enough reserved energy to move the broken pipe end)</td>
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<td>9SI621</td>
<td>BT-SI-03</td>
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<td>PW No PW occurs.(not enough reserved energy to move the broken pipe end)</td>
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<td>9SI622</td>
<td>BT-SI-04</td>
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<td>Figure 3.6-39  (1 of 2)</td>
<td>PW No PW occurs.(not enough reserved energy to move the broken pipe end)</td>
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<td>Break Type</td>
<td>Break Location</td>
<td>Protection of Essential Target</td>
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<td>9SI611</td>
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<td>PW El. 136’-6” floor (SCH41-01) (designed to withstand PW impact load)</td>
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<td>JI El. 136’-6” floor (SCH41-01) (designed to withstand JI load)</td>
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<td>BT-SI-06</td>
<td>C</td>
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<td>PW El. 136’-6” floor (SCH21-01) (designed to withstand PW impact load)</td>
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<td>JI El. 136’-6” floor (SCH21-01) (designed to withstand JI load)</td>
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<td>PW El. 136’-6” floor (SCH11-01) (designed to withstand PW impact load)</td>
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<td>JI El. 136’-6” floor (SCH11-01) (designed to withstand JI load)</td>
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<td>9SI614</td>
<td>BT-SI-08</td>
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<td>PW El. 136’-6” floor (SCH31-01) (designed to withstand PW impact load)</td>
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<td>JI El. 136’-6” floor (SCH31-01) (designed to withstand JI load)</td>
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Chemical and Volume Control System (Not applied for Leak-Before-Break): Terminal End Break

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<th>Break Type</th>
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<td>9RC105</td>
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<td>Figure 3.6-40</td>
<td>PW Regenerative heat exchanger room floor (SCE31-01) (designed to withstand PW impact load)</td>
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<td>JI Regenerative heat exchanger (designed to withstand JI load)</td>
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<td>9CV106</td>
<td>BT-CV-02</td>
<td>C</td>
<td>Figure 3.6-40</td>
<td>PW Regenerative heat exchanger room floor (SCE31-01) (designed to withstand PW impact load)</td>
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<td>JI Regenerative heat exchanger (designed to withstand JI load)</td>
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<td>9CV102</td>
<td>BT-CV-03</td>
<td>C</td>
<td>Figure 3.6-40</td>
<td>PW Regenerative heat exchanger room wall (WCM11-03) (designed to withstand PW impact load)</td>
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<td>JI Regenerative heat exchanger (designed to withstand JI load)</td>
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<td>BT-CV-04</td>
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<td>PW Regenerative heat exchanger floor (SCE31-01) (designed to withstand PW impact load)</td>
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Table 3.6-9 (4 of 8)

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<th>Subsystem</th>
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<th>Break Type</th>
<th>Break Location</th>
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<td>9CV106</td>
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<td>PW Letdown heat exchanger room wall (WCE11-01) (designed to withstand PW impact load)</td>
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<td>JI Letdown heat exchanger, letdown heat exchanger wall (WCE11-01) (designed to withstand JI load)</td>
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<tr>
<td>9CV107</td>
<td>BT-CV-06</td>
<td>C</td>
<td>Figure 3.6-40 (1 of 2)</td>
<td>PW Letdown heat exchanger room wall (WCM11-01), CC line (CC094A8) (designed to withstand PW impact load)</td>
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<td>JI Letdown heat exchanger, letdown heat exchanger room floor (SCH11-01)(designed to withstand JI load)</td>
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<td>BT-CV-07</td>
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<td>C</td>
<td>Figure 3.6-40 (1 of 2)</td>
<td>PW No PW occurs. (impacted piping SI008EB12 has larger nominal size and thickness than whipping pipe)</td>
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<td>JI Containment wall, containment penetration (PC0402) (designed to withstand JI load)</td>
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<td>BT-CV-08</td>
<td>C</td>
<td>Figure 3.6-40 (1 of 2)</td>
<td>PW IRWST venting opening (designed to withstand PW impact load)</td>
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<td>JI Containment wall, containment penetration (PC0230) (designed to withstand JI load)</td>
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<tr>
<td>9CV622</td>
<td>BT-CV-09</td>
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<td>Figure 3.6-40 (2 of 2)</td>
<td>PW No essential target</td>
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<td>JI Containment wall, containment penetration (PC0304) (designed to withstand JI load)</td>
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<tr>
<td>9CV101</td>
<td>BT-CV-10</td>
<td>C</td>
<td>Figure 3.6-40 (1 of 2)</td>
<td>PW No PW occurs because of geometric configuration (broken pipe is restrained by wall)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
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<td>JI No essential target</td>
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Steam Generator Blowdown System (Not applied for Leak-Before-Break): Terminal End Break

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<th>Subsystem</th>
<th>Break Number</th>
<th>Break Type</th>
<th>Break Location</th>
<th>Protection of Essential Target</th>
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<tbody>
<tr>
<td>9SD103</td>
<td>BT-SD-01</td>
<td>C</td>
<td>Figure 3.6-41 (1 of 1)</td>
<td>PW PWR (HSD103-001W) installed to protect RCP01A</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>JI SG1 (designed to withstand JI load)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Blast RCP #1A (designed to withstand blast load)</td>
</tr>
<tr>
<td>BT-SD-02</td>
<td></td>
<td>C</td>
<td>Figure 3.6-41 (1 of 1)</td>
<td>PW PWR (HSD103-002W) required to protect RCP01B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>JI SG1 (designed to withstand JI load)</td>
</tr>
<tr>
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<td>Blast RCP #1B (designed to withstand blast load)</td>
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### Table 3.6-9 (5 of 8)

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<td>PW PWR (HSD104-001W) required to protect RCP02A</td>
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<td>JI SG2 (designed to withstand JI load)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Blast RCP #2A (designed to withstand blast load)</td>
</tr>
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<td>BT-SD-04</td>
<td>C</td>
<td>Figure 3.6-41</td>
<td>PW PWR (HSD104-002W) required to protect RCP02B</td>
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<td></td>
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<td>(1 of 1)</td>
<td>JI SG2 (designed to withstand JI load)</td>
</tr>
<tr>
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<td></td>
<td></td>
<td></td>
<td>Blast RCP #2B (designed to withstand blast load)</td>
</tr>
<tr>
<td>9SD103</td>
<td>BT-SD-05</td>
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<td>Figure 3.6-41</td>
<td>PW PWR (HSD103-003W) required to protect containment liner and to restrain nozzle load due to PW</td>
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<tr>
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<td></td>
<td></td>
<td>(1 of 1)</td>
<td>JI Containment Penetration(PC0911) (designed to withstand JI load)</td>
</tr>
<tr>
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<td>Blast No essential target other than Containment Penetration(PC0911) (bounded by JI load)</td>
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<td>9SD104</td>
<td>BT-SD-06</td>
<td>C</td>
<td>Figure 3.6-41</td>
<td>PW PWR (HSD104-003W) required to protect containment liner and to restrain nozzle load due to PW</td>
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<td>JI Containment Penetration(PC0912) (designed to withstand JI load)</td>
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<td>BT-SD-07</td>
<td>C</td>
<td>Figure 3.6-41</td>
<td>PW Secondary shield wall (WCE21-01) (designed to withstand PW impact load)</td>
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<tr>
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<td>(1 of 1)</td>
<td>JI Secondary shield wall (WCE21-01) (designed to withstand JI load)</td>
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<td>Blast Secondary shield wall (WCE21-01) (designed to withstand blast load)</td>
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<td>9SD102</td>
<td>BT-SD-08</td>
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<td>PW Secondary shield wall (WCE11-01) (designed to withstand PW impact load)</td>
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<td>JI Secondary shield wall (WCE11-01) (designed to withstand JI load)</td>
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<td>Blast Secondary shield wall (WCE11-01) (designed to withstand blast load)</td>
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### Table 3.6-9 (6 of 8)

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<th>Break Type</th>
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<tr>
<td><strong>Main Steam System (Not applied for Leak-Before-Break): Terminal End Break</strong></td>
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<tr>
<td>9MS104 9MS272 BT-MS-01 C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW</td>
<td>PWR (HMS104-001W) installed (to protect containment wall (liner plate))</td>
</tr>
<tr>
<td></td>
<td></td>
<td>JI</td>
<td>Top of SG1 (designed to withstand JI load)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Blast</td>
<td>SG1 (designed to withstand blast load)</td>
</tr>
<tr>
<td>BT-MS-02 C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW</td>
<td>PWR(HMS104-002W) installed (to protect feedwater pipe, SG1 nozzle)</td>
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<tr>
<td></td>
<td>JI</td>
<td>Containment penetration (PC0611) (designed to withstand JI load)</td>
<td></td>
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<tr>
<td></td>
<td>Blast</td>
<td>FW006 (impulse loading only) CS011,CS003 (out of graded approach scope)</td>
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<tr>
<td>9MS103 9MS271 BT-MS-03 C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW</td>
<td>PWR(HMS103-001W) installed (to protect containment wall (liner plate))</td>
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<tr>
<td></td>
<td>JI</td>
<td>Top of SG1 (designed to withstand JI load)</td>
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<tr>
<td></td>
<td>Blast</td>
<td>SG1 designed to withstand blast load</td>
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<tr>
<td>BT-MS-04 C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW</td>
<td>PWR(HMS103-002W) installed (to protect feedwater pipe, SG1 nozzle)</td>
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<td>JI</td>
<td>Containment penetration (PC612) (designed to withstand JI load)</td>
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<td>Blast</td>
<td>FW006 (impulse loading only) CS011,CS003 (out of graded approach scope)</td>
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<td>9MS102 9MS270 BT-MS-05 C</td>
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<td>PW</td>
<td>PWR (HMS102-001W) installed (to protect containment wall (liner plate))</td>
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<td></td>
<td>JI</td>
<td>Top of SG2 (designed to withstand JI load)</td>
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<td>Blast</td>
<td>SG2 (designed to withstand blast load)</td>
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<tr>
<td>BT-MS-06 C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW</td>
<td>PWR (HMS102-002W) installed (to protect feedwater pipe (FW045AB14), SG2 nozzle)</td>
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<td>JI</td>
<td>Containment penetration (PC622) (designed to withstand JI load)</td>
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<td>Blast</td>
<td>FW006 (impulse loading only) CS011,CS003 (out of graded approach scope)</td>
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### Table 3.6-9 (7 of 8)

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<th>Subsystem</th>
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<th>Break Type</th>
<th>Break Location</th>
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<tr>
<td><strong>Main Steam System</strong> Not applicable Leak-Before-Break: Terminal End Break</td>
<td></td>
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<tr>
<td>9MS101</td>
<td>BT-MS-07</td>
<td>C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW PWR (HMS101-001W) installed (to protect containment wall (liner plate))</td>
</tr>
<tr>
<td>9MS269</td>
<td></td>
<td></td>
<td></td>
<td>JI Top of SG2 (designed to withstand JI load)</td>
</tr>
<tr>
<td></td>
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<td>Blast SG2 (designed to withstand blast load)</td>
</tr>
<tr>
<td></td>
<td>BT-MS-08</td>
<td>C</td>
<td>Figure 3.6-42 (1 of 1)</td>
<td>PW PWR (HMS101-002W) installed (to protect feedwater pipe, SG2 nozzle)</td>
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<td>JI Containment penetration (PC621) (designed to withstand JI load)</td>
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<td>Blast FW006 (impulse loading only) CS011,CS003 (out of graded approach scope)</td>
</tr>
<tr>
<td><strong>Feedwater System</strong> (Not applied for Leak-Before-Break): Terminal End Break</td>
<td></td>
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<td></td>
</tr>
<tr>
<td>9FW101</td>
<td>BT-FW-01</td>
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<td>Figure 3.6-43 (1 of 1)</td>
<td>PW PWR (HFW101-001W) installed (to minimize load to SG1 nozzle)</td>
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<tr>
<td>9FW209</td>
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<td>JI Containment penetration (PC0511) (designed to withstand JI load)</td>
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<tr>
<td>9FW102</td>
<td>BT-FW-02</td>
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<td>PW PWR (HFW102-001W) installed to minimize load to SG2 nozzle</td>
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<td>JI Containment penetration (PC0521) (designed to withstand JI load)</td>
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<td>9FW101</td>
<td>BT-FW-03</td>
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<td>Figure 3.6-43 (1 of 1)</td>
<td>PW Secondary shield wall (WCE41-01) (designed to withstand PW impact load)</td>
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<td>JI Secondary shield wall (WCE41-01), SG1 (designed to withstand JI load)</td>
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<td>9FW101</td>
<td>BT-FW-04</td>
<td>C</td>
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<td>PW Secondary shield wall (WCE21-01) (designed to withstand PW impact load)</td>
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<td>JI Secondary shield wall (WCE21-01), SG2 (designed to withstand JI load)</td>
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<td>9FW101</td>
<td>BT-FW-05</td>
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<td>Figure 3.6-43 (1 of 1)</td>
<td>PW Secondary shield wall (WCE31-01) (designed to withstand PW impact load)</td>
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<td>JI Secondary shield wall (WCE31-01), SG1 (designed to withstand JI load)</td>
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<td>Subsystem</td>
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<td>Break Type</td>
<td>Break Location</td>
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<td>BT-FW-06</td>
<td>C</td>
<td>Figure 3.6-43 (1 of 1)</td>
<td>PW: PWR (HFW102-002W) installed to minimize load to pressurizer spray piping, RC007C4. JI: Secondary shield wall (WCE11-01), SG2 (designed to withstand JI load).</td>
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<td>9AF101</td>
<td>BT-FW-07</td>
<td>C</td>
<td>Figure 3.6-43 (1 of 1)</td>
<td>PW: PWR (HAF101-001W) required to minimize load to SG1 nozzle. JI: Containment penetration (PC0512), PWR (HAF101-001W) (designed to withstand JI load).</td>
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<td>9AF102</td>
<td>BT-FW-08</td>
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<td>Figure 3.6-43 (1 of 1)</td>
<td>PW: PWR (HFW102-001W) installed to minimize load to SG1 nozzle. JI: Containment penetration (PC0522), PWR (HAF102-001W) (designed to withstand JI load).</td>
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<td>9AF101</td>
<td>BT-FW-09</td>
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<td>Figure 3.6-43 (1 of 1)</td>
<td>PW: Secondary shield wall (WCE21-01) (designed to withstand PW load). JI: Secondary shield wall (WCE21-01), SG1 (designed to withstand JI load).</td>
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<td>BT-FW-10</td>
<td>C</td>
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<td>PW: Secondary shield wall (WCE11-01) (designed to withstand PW load). JI: Secondary shield wall SG2 (WCE11-01) (designed to withstand JI load).</td>
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<td>9FW601</td>
<td>BT-FW-11</td>
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<td>PW: Steel structure (3CS502) (designed to withstand PW load). JI: No essential target.</td>
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<tr>
<td>9FW627</td>
<td>BT-FW-12</td>
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<td>PW: No essential target. JI: No essential target.</td>
</tr>
<tr>
<td>9FW602</td>
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<td>PW: No essential target. JI: No essential target.</td>
</tr>
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<td>9FW628</td>
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<td>PW: Steel structure (3CS501) (designed to withstand PW load). JI: No essential target.</td>
</tr>
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Figure 3.6-1 Typical Pipe Whip Restraint Configuration
Figure 3.6-2  LBB Design Criteria Development Diagram
Figure 3.6-3 Stress-Strain Curve for Hot Leg and Cold Leg for Leakage Crack Length Calculation
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Figure 3.6-4  Stress-Strain Curve for Surge Line for Leakage Crack Length Calculation
Figure 3.6-5  Stress-Strain Curve for DVI (Group 1) for Leakage Crack Length Calculation
Figure 3.6-6  Stress-Strain Curve for DVI/SC (Group 2&3) for Leakage Crack Length Calculation
Figure 3.6-7  Stress-Strain Curve for Hot Leg and Cold Leg for Crack
Stability Analysis
Figure 3.6-8  Stress-Strain Curve for Surge Line for Crack Stability Analysis
Figure 3.6-9  Stress-Strain Curve for DVI (Group 1)
Figure 3.6-10  Stress-Strain Curve for DVI/SC (Group 2&3)
Figure 3.6-11  J-R Curve for Hot Leg and Cold Leg
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Figure 3.6-12  J-R Curve for Surge Line
Figure 3.6-13  J-R Curve for DVI (Group 1)
Figure 3.6-14  J-R Curve for DVI/SC (Group 2&3)
Figure 3.6-15  Moment vs. Crack Length for Surge Line (Hot leg nozzle/Pipe Interface)
Figure 3.6-16  Moment vs. Crack Length for Surge Line (Intermediate Pipe)
Figure 3.6-17  Moment vs. Crack Length for Surge Line  
(Pressurizer nozzle/Pipe Interface)
Figure 3.6-18  Moment vs. Crack Length for Cold Leg
Figure 3.6-20  Moment vs. Crack Length for DVI (Group 1)
Figure 3.6-21  Moment vs. Crack Length for DVI (Group 2)
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Figure 3.6-22  Moment vs. Crack Length for SC (Group 3)
Figure 3.6-23  Overall Finite Element Model

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Figure 3.6-24  Crack Area Close-up of Finite Element Model
Figure 3.6-25  Different Crack Length Used to Calculate Derivative

\[ l = \text{Leakage crack length} \]
\[ a = \text{Model crack length} \]
\[ a \pm d = \text{Model crack length} \pm \text{a small amount} \]
Figure 3.6-26 Development of a Stability Evaluation Diagram
Figure 3.6-27  LBB Piping Evaluation Diagram
Figure 3.6-28  Use of the LBB Piping Evaluation Diagram

- Point 1 passes LBB on ‘a’ and ‘2a’
- Point 2 fails maximum moment on ‘2a’
- Point 11 fails $\sqrt{2}$ (maximum moment) on ‘a’
- Point 14 fails $\sqrt{2}$ (maximum moment) on ‘a’ and maximum moment on ‘2a’
Figure 3.6-29  LBB Piping Evaluation Diagram (Hot Leg)
Figure 3.6-30  LBB Piping Evaluation Diagram (Cold Leg)
Figure 3.6-31  LBB Piping Evaluation Diagram (Surge Line-Hot leg nozzle/ Pipe Interface)
Figure 3.6-32  LBB Piping Evaluation Diagram (Surge Line-Intermediate Pipe)
Figure 3.6-33  LBB Piping Evaluation Diagram (Surge Line-Pressurizer nozzle/Pipe Interface)
Figure 3.6-34  LBB Piping Evaluation Diagram (DVI Group 1)
Figure 3.6-35  LBB Piping Evaluation Diagram (DVI Group 2)
Figure 3.6-36  LBB Piping Evaluation Diagram (SC Group 3)
Figure 3.6-37  High-Energy Piping Portion and Break Location for Reactor Coolant System (1 of 2)

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Figure 3.6-37  High-Energy Piping Portion and Break Location for Reactor Coolant System (2 of 2)
Figure 3.6-38  High-Energy Piping Portion and Break Location for Reactor Coolant Gas Vent System
Figure 3.6-39  High-Energy Piping Portion and Break Location for Safety Injection and Shutdown Cooling System (1 of 2)

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Figure 3.6-39  High-Energy Piping Portion and Break Location for Safety Injection and Shutdown Cooling System (2 of 2)
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Figure 3.6-40  High-Energy Piping Portion and Break Location for Chemical and Volume Control System (1 of 2)
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Figure 3.6-40 High-Energy Piping Portion and Break Location for Chemical and Volume Control System (2 of 2)
Figure 3.6-41  High-Energy Piping Portion and Break Location for Steam Generator Blowdown System
Figure 3.6-42  High-Energy Piping Portion and Break Location for Main Steam System
Figure 3.6-43  High-Energy Piping Portion and Break Location for Feedwater System
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Figure 3.6-44 High-Energy Piping Portion and Break Location for Auxiliary Feedwater System
3.7 Seismic Design

The APR1400 structures, systems, and components (SSCs) important to safety are designed to withstand the effects of earthquakes without loss of capability to perform their safety functions, as required by 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2 (Reference 1).

The APR1400 SSCs are classified in one of the following three seismic categories:

a. Seismic Category I
b. Seismic Category II
c. Seismic Category III (non-seismic)

Seismic Category I SSCs are designed to withstand the effects of the earthquake event and to maintain their specified design functions. Seismic Category II SSCs do not perform safety-related functions, but structural failure or interaction could degrade the function of a seismic Category I SSC to an unacceptable safety level. Seismic Category III SSCs do not perform safety-related functions, and structural failure or interaction could not degrade the function of a seismic Category I SSC to an unacceptable safety level.

3.7.1 Seismic Design Parameters

This subsection describes seismic design parameters such as design ground motions and damping values that are used in the design of the SSCs important to safety and classified as seismic Category I in Section 3.2.

The APR1400 seismic Category I SSCs are designed for the safe shutdown earthquake (SSE). The SSE is defined as the maximum potential vibratory ground motion at the generic plant site. Since the operating basis earthquake (OBE) is defined as one third the SSE, an explicit analysis or design of the APR1400 seismic Category I SSCs based on OBE is not required in accordance with Appendix S of 10 CFR Part 50 (Reference 2).

3.7.1.1 Design Ground Motion

The design response spectra of the site-independent SSE are now referred to as the certified seismic design response spectra (CSDRS). The CSDRS and design time histories compatible with CSDRS are described in the following subsections.
3.7.1.1 Design Ground Motion Response Spectra

The peak ground acceleration (PGA) of the CSDRS has been established as 0.3g for the APR1400 design for both the horizontal and vertical directions.

The horizontal and vertical CSDRS for the APR1400 are based on the NRC Regulatory Guide (RG) 1.60 (Reference 3) response spectra, enriched in the high frequency range in the following manner.

a. The spectral amplitudes of the horizontal and vertical response spectra at control frequencies 9 Hz and below are equal to those of the NRC RG 1.60 response spectra.

b. The control frequency at which the PGA is reached is changed from 33 Hz to 50 Hz for both the horizontal and vertical spectra.

c. A control frequency at 25 Hz is added. The spectral amplitudes at 25 Hz are set to the NRC RG 1.60 response spectra at 25 Hz scaled by a factor of 1.30 for both the horizontal and vertical spectra.

d. Linearly vary the modified spectra, on a log-log-scale, between the control frequencies 9 Hz, 25 Hz, and 50 Hz.

The CSDRS at damping values of 3 and 4 percent are developed from the CSDRS at damping values of 2 and 5 percent by linear interpolation using ASCE 4, Section 2.2.1, Equation 2.2-1.

The digitized values of the resulting APR1400 horizontal and vertical CSDRS for 2, 3, 4, 5, 7, and 10 percent damping values are provided in Table 3.7-1. The APR1400 horizontal and vertical CSDRS are presented in Figures 3.7-1 and 3.7-2, respectively.

The CSDRS are applied at the finished grade in the free-field as an additional requirement from 10 CFR Part 50, Appendix S. Figures 3.7A-9 (1 of 3 and 2 of 3), 3.7A-10 (1 of 3 and 2 of 3), and 3.7A-11 (1 of 3 and 2 of 3) in Appendix 3.7A show that the horizontal components of CSDRS in the free-field at the foundation level of the APR1400 standard plant seismic Category I structures satisfy the PGA of at least 0.1g. The vertical component of CSDRS in the free-field at the foundation level of the APR1400 standard plant seismic Category I structures is presented in Figures 3.7A-9 (3 of 3), 3.7A-10 (3 of 3), 3.7A-11 (3 of 3), 3.7A-12 (3 of 3), and 3.7A-13 (3 of 3).
and 3.7A-11 (3 of 3) in Appendix 3.7A. The site response motions in Figure 3.7A-9 through 3.7A-11 correspond to outcrop motion in the free-field.

The site-specific seismic design can be developed for other seismic Category I and II SSCs, which are not included in the APR1400 standard plant design, at the combined license (COL) stage using the site-specific SSE derived from the ground motion response spectra (GMRS) in accordance with NRC RG 1.208 (Reference 4).

3.7.1.1.2 Design Ground Motion Time History

The three design acceleration time histories composed of two horizontal (H1 and H2) and one vertical components (VT), which envelop the CSDRS, are applied in both soil-structure interaction analyses and fixed-base analyses of seismic Category I structures. The initial seed motions that were modified to create the design time histories are actual seed-recorded Northridge earthquake time histories.

The design time histories are generated with an increment of time size of 0.005 second to provide a Nyquist frequency of 100 Hz. Figures 3.7-3, 3.7-4, and 3.7-5 show the acceleration, velocity, and displacement time histories for H1, H2, and VT components for each time step, respectively. The design time histories, H1, H2, and VT, are applied in the east-west (E-W) direction, north-south (N-S) direction, and vertical direction, respectively. The absolute values of correlation coefficients for each pair of the design time histories are as follows:

Correlation coefficient for H1 and H2 = 0.032

Correlation coefficient for H1 and VT = 0.079

Correlation coefficient for H2 and VT = 0.029

The design time histories are statistically independent because the correlation coefficients between the design time histories are less than 0.16 as specified in Standard Review Plan (SRP) 3.7.1 (Reference 5). Therefore, the representative maximum response of interest of the APR1400 SSCs can be obtained either by performing separate analyses for each of the three components of design time histories or by performing a single analysis with all three components of design time histories applied simultaneously.
The design time histories have a total time duration equal to 20.48 seconds and a corresponding stationary phase, which is the strong-motion duration defined as the time required for the Arias Intensity rise from 5 percent to 75 percent in more than 6 seconds.

The design time histories are developed following the spectrum matching acceptance criteria of Option 1, Approach 1, in Section II of SRP 3.7.1. The comparison plots of the response spectra of the design time histories versus the design response spectra for 2, 3, 4, 5, 7, and 10 percent critical damping are shown in Figures 3.7-6, 3.7-7, and 3.7-8. The figures demonstrate that the design time histories envelop the design response spectra for those damping values, satisfying the requirement of SRP 3.7.1 that no more than 5 points fall below and by no more than 10 percent below the design response spectra. The response spectra are computed at the frequency intervals given in Table 3.7.1-1 of SRP 3.7.1.

In SRP 3.7.1, the requirement of minimum power spectral density (PSD) to prevent the design ground acceleration time histories from having a deficiency of power over any frequency range is described. SRP 3.7.1 specifies that the use of a single time history is justified by satisfying a target PSD requirement in addition to the design response spectra enveloping requirements.

Since the original NRC RG 1.60 horizontal spectrum and the horizontal CSSDRS are identical for frequencies less than 9 Hz, no modification to the target horizontal PSD is done in this frequency range. This approach is closely matched as compared to the target PSD function developed by using the method described in NUREG/CR-5347 for the frequency range below 9 Hz.

The time-history simulation method described in NUREG/CR-5347 (Reference 8) is used to develop the CSSDRS-compatible horizontal target PSD in the higher frequency range above 9 Hz. The time-history PSDs for the 30 time histories of the CSSDRS compatible time-history ensembles are calculated and their ensemble-averaged PSDs are computed. After smoothing of the ensemble-averaged PSDs calculated in accordance with NUREG/CR-5347, the target PSDs compatible with the CSSDRS are obtained. The smoothed ensemble-mean PSDs obtained from the PSDs of the 30 time history ensembles are segmentally smoothed using piecewise log-log linear. The detailed procedure for generating target PSD is described in Technical Report, APR1400-E-S-NR-14001-P. The resulting piecewise log-log linear horizontal target PSD developed is given in Table 3.7-3. The minimum required horizontal PSD is then 0.8 times the horizontal target PSD.
The vertical target PSD, compatible with the vertical CSDRS, is obtained from the horizontal target PSD, compatible with the horizontal CSDRS using the following equation:

$$S_V(f) = \left[ \frac{R_V(f, 2\%)}{R_H(f, 2\%)} \right]^2 \times S_H(f)$$

where $R_H(f, 2\%)$ and $R_V(f, 2\%)$ are, respectively, the 2 percent damped horizontal and vertical CSDRS values at the frequency ($f$). The detailed procedure for generating target PSD is described in Technical Report, APR1400-E-S-NR-14001-P (Reference 9). The minimum required vertical PSD is then 0.8 times the vertical target PSD.

The PSDs of the design acceleration time histories are presented in Figures 3.7-9 through 3.7-11. The PSDs of the design acceleration time histories exceed the minimum required PSD throughout the entire frequency range. The PSDs presented are the averaged PSDs obtained over a moving frequent band of ± 20 percent centered at each frequency. The PSD amplitude at frequency ($f$) has the averaged PSD amplitude between the frequency range of $0.8f$ and $1.2f$ as stated in Appendix A of SRP 3.7.1.

The PSDs of the design acceleration time histories in accordance with SRP 3.7.1, Appendix B are computed to further demonstrate satisfying the PSD requirement. As shown in Figure 3.7-9 (b) through 3.7-11 (b), the PSDs are reasonable compared with the PSDs consistent with SRP 3.7.1, Appendix B.

3.7.1.1.3 Hard Rock High Frequency Seismic Input Motions

GMRS for some Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS. The GMRS for such a site are called hard rock high frequency (HRHF) seismic input motions. The hard rock is defined as having the low-strain shear wave velocity greater than 2,084 m/sec (9,200 ft/sec) as described in Subsection 2.5.2.6. The PGA of the HRHF target response spectra are prescribed as 0.46g for the evaluation of the APR1400 standard plant design for both the horizontal and vertical directions. The HRHF horizontal and vertical target response spectra are shown in Figures 3.7-12 and 3.7-13, respectively. These HRHF response spectra exceed the CSDRS for frequencies higher than approximately 10 Hz.

The three acceleration time histories composed of two horizontal (H1H and H2H) and one vertical component (VTH), which envelop the HRHF response spectra, are generated. The time histories, H1H, H2H, and VTH are applied in the E-W direction, N-S direction, and vertical direction, respectively. The initial seed motions that were modified to create
the time histories compatible with HRHF response spectra are actual seed-recorded Nahanni earthquake time histories.

The time histories are generated with an increment of time size of 0.005 second. Figures 3.7-14, 3.7-15, and 3.7-16 show the acceleration, velocity, and displacement time histories for H1H, H2H, and VTH components for each time step, respectively. The absolute values of correlation coefficients for each pair of the time histories are as follows:

Correlation coefficient for H1H and H2H = 0.028

Correlation coefficient for H1H and VTH = 0.031

Correlation coefficient for H2H and VTH = 0.036

The time histories are statistically independent because the correlation coefficients between the design time histories are less than 0.16. The time histories have a total time duration equal to 20.48 seconds and a corresponding stationary phase, which is the strong-motion duration defined as the time required for the Arias Intensity rise from 5 percent to 75 percent in more than 6 seconds.

The time histories are developed following the spectrum matching acceptance criteria of Option 1, Approach 1, in Section II of SRP 3.7.1. The comparison plots of the response spectra of the time histories versus the HRHF response spectra for 2, 3, 4, 5, 7, and 10 percent critical dampings are shown in Figures 3.7-17, 3.7-18, and 3.7-19. The figures demonstrate that the time histories envelop the HRHF response spectra for those damping values, satisfying the requirement of SRP 3.7.1 that no more than 5 points fall below and by no more than 10 percent below the HRHF response spectra.

The PSDs of the acceleration time histories compatible with the HRHF response spectra in accordance with SRP 3.7.1, Appendix B are computed to further demonstrate satisfying the PSD requirement. As shown in Figure 3.7-20 (b) through 3.7-22(b), the PSDs are reasonable compared with the PSDs consistent with SRP 3.7.1, Appendix B.

For the development of the HRHF-response spectra-compatible target PSDs in the frequency range from 0.3 to 100 Hz, the time-history simulation method described in NUREG/CR-5347 is used. The resulting piecewise log-log linear horizontal and vertical target PSD developed is given in Tables 3.7-5 and 3.7-6. The minimum required horizontal and vertical PSD is then 0.8 times the horizontal and vertical target PSD.
The PSDs of the acceleration time histories compatible with the HRHF response spectra are presented in Figures 3.7-20 through 3.7-22. The PSDs of the acceleration time histories exceed the minimum required PSD throughout the entire frequency range.

The evaluation methodology and results of the APR1400 for the HRHF seismic input motions are provided in Appendix 3.7B.

3.7.1.2 Percentage of Critical Damping Values

Damping values used for various nuclear safety-related SSCs are based on NRC RG 1.61 (Reference 10). These values are expressed in percentages of critical damping and are given in Table 3.7-7. Damping values of soil to be used in soil-structure interaction analysis are provided in Tables 3.7A-1 through 3.7A-8.

3.7.1.3 Supporting Media for Seismic Category I Structures

Seismic Category I structures are founded directly on rock or competent soil. The nuclear island and emergency diesel generator building correspond to the seismic Category I structures of the APR1400 standard plant design. The nuclear island consists of the following seismic Category I structures, the reactor containment building and the auxiliary building, which are founded on a common basemat. The emergency diesel generator building and a diesel fuel oil storage tank room are also seismic Category I structures. The foundation embedment depth, foundation size, and total height of the seismic Category I structures are presented in Table 3.7-8.

For the design of seismic Category I structures, eight soil profiles and one fixed-base condition are established with various shear wave velocities compared with soil depth.

The supporting media for the generic site are described in Appendix 3.7A about soil properties, layering characteristics, shear wave velocity, shear modulus, and density. The site-specific soil degradation models can be determined from dynamic laboratory testing of the site materials or from the published literature. The COL applicant is to demonstrate the applicability of soil degradation models used in site-specific site response analysis for the site conditions (COL 3.7(1)). The COL applicant is to compare the site-specific strain-compatible soil properties with generic soil properties in order to confirm that the site meets the generic soil profile used in the standard design (COL 3.7(2)).
These eight profiles are considered representative to envelop sites where competent soil is defined by the shear wave velocity of the supporting medium at the foundation level exceeding 304.8 m/sec (1,000 ft/sec). The shear wave velocity profiles of the eight sites considered are shown in Figure 3.7-23. The eight soil profiles, S1 through S4 and S6 through S9, are developed as combinations of six soil layering categories, which are designated as 55, 100, 200, 500, 1,000 ft, and half-space, and five shear-wave-velocity categories, namely, 1,200, 2,000, 4,000, 6,000, and 9,200 ft/sec. The generic site soil profiles are described further in Technical Report, APR1400-E-S-NR-14001-P (Reference 9).

To generate more conservative seismic responses, the extreme groundwater level (El. 98 ft 8 in.) in Subsection 3.8A.1.4.2.3.2 is considered in the seismic analyses of the seismic Category I structures, rather than the design groundwater level (El. 96 ft 8 in.) in Subsection 3.8.4.3.1.

3.7.2 Seismic System Analysis

This subsection describes the seismic analysis methods and models for the seismic Category I structures of the APR1400 standard plant design such as the reactor containment building, auxiliary building and emergency diesel generator building. The three-dimensional finite element models (FEMs) having an adequate number of discrete mass degrees of freedom are developed to capture the global and local translational, rocking, and torsional responses of the structures.

The COL applicant is to provide the seismic design of seismic Category I SSCs and seismic Category II structures that are not part of the APR1400 standard plant design. The seismic Category I and II structures are as follows (COL 3.7(3)):

a. Seismic Category I essential service water building
b. Seismic Category I component cooling water heat exchanger building
c. Seismic Category II turbine generator building
d. Seismic Category II compound building
e. Seismic Category II alternate alternating current gas turbine generator building
3.7.2.1 Seismic Analysis Methods

Seismic Category I SSCs are identified in Section 3.2. The safety-related structures are modeled as three-dimensional FEMs for the seismic analysis. Table 3.7-9 summarizes the types of models, computer programs, and analysis methods that are used in the seismic analyses of the seismic Category I structures, as well as the purposes of the dynamic analyses. Figures 3.7-24, 3.7-25, 3.7-26, and 3.7-27 show FEMs of the safety-related structures for the reactor containment building containment structure, reactor containment building internal structure, auxiliary building, and emergency diesel generator building.

Further details of dynamic modeling of building structures for seismic analysis are described in Subsection 3.7.2.3. The structural model is analyzed with a separate consideration of the excitation in each of the three orthogonal directions, E-W, N-S, and vertical. The results are then combined as described in Subsection 3.7.2.6. The seismic analysis of the above structures is performed by one of the following methods described below.

3.7.2.1.1 Response Spectrum Analysis

The response of a multi-degree-of-freedom system subjected to seismic excitation is represented by the following equation of motion:

\[ [M] \{\ddot{X}\} + [C] \{\dot{X}\} + [K] \{X\} = 0 \]

Where:

\[ [M] = \text{mass matrix} \ (n \times n) \]
\[ [C] = \text{damping matrix} \ (n \times n) \]
\[ [K] = \text{stiffness matrix} \ (n \times n) \]
\[ \{X\} = \text{column vector of relative displacements} \ (n \times 1) \]
\[ \{\dot{X}\} = \text{column vector of relative velocities} \ (n \times 1) \]
\[ \{\ddot{X}\} = \text{column vector of relative accelerations} \ (n \times 1) \]
\[ n = \text{number of dynamic degrees of freedom} \]
\[
\{ \dot{U}_g \} = \text{column vector of ground accelerations } (n \times 1)
\]

In the response spectrum analysis, the equations of motion are decoupled using the transformation:

\[
\{ X \} = \{ \varphi \} \{ Y \}
\]

Where:

\[
\{ \varphi \} = \text{mode shape matrix}
\]

\[
\{ Y \} = \text{vector of normal or generalized coordinates } (m \times 1)
\]

\[
m = \text{number of modes considered}
\]

The decoupled equation of motion for each mode is transformed to a single degree of freedom system:

\[
\ddot{Y}_j + 2 \lambda_j \omega_j \dot{Y}_j + \omega_j^2 Y_j = -\Gamma_j \dot{U}_g
\]

Where:

\[
Y_j = \text{generalized coordinate of the jth mode}
\]

\[
\lambda_j = \text{damping ratio for the jth mode expressed as fraction of critical damping}
\]

\[
\omega_j = \text{circular frequency of the jth mode of the system}
\]

\[
\Gamma_j = \text{modal participation factor of the jth mode} = \frac{\{ \varphi_j \}^T \{ M \} \{ 1 \}}{\{ \varphi_j \}^T \{ M \} \{ \varphi_j \}}
\]

The generalized maximum response of each mode is determined from:

\[
Y_j(\text{max}) = \Gamma_j \frac{S_{aj}}{\omega_j}
\]

Where:

\[
S_{aj} \text{ is the spectral acceleration corresponding to frequency } \omega_j.
\]
The maximum displacement at node $i$ relative to the base due to mode $j$ is:

$$X_{ij}(max) = \phi_{ij} Y_j(max)$$

The modal response $X_{ij}(max)$ is used to determine other modal response quantities, such as forces. The modal combination method is used to obtain the final response by the methods described in Subsection 3.7.2.7.

Response spectrum analysis is used in Subsection 3.8.1.4.4, 3.8.3.4.1, and Appendix 3.8A to compute only the seismic design forces of the containment structure and internal structure in the reactor containment building using the in-structure response spectra at the top of basemat generated from seismic soil-structure interaction analysis. The seismic response forces obtained from the response spectrum analysis are then combined with other design loads to design structural members of the containment structure and internal structure.

3.7.2.1.2 Time-History Methods

The solution of the equation of motion given in Subsection 3.7.2.1.1 is obtained using one of three methods: modal superposition, direct integration, or complex frequency response in the frequency domain.

The method utilizes mode superposition or direct integration for time-history analysis and is used as an alternative analysis option for seismic Category I systems and subsystems. The seismic responses of the systems and subsystems that are seismic Category I SSCs are obtained using the finite element method. The analyses of all of the systems are performed for three orthogonal (two horizontal and one vertical) components of in-structure response time histories at the points of attachment.

**Modal Superposition Method**

The modal superposition method is used when the equations of motion can be decoupled as given in Subsection 3.7.2.1.1. Then the decoupled equation of motion for each mode is integrated using a proven technique, such as those listed in Table 3.2-1 of ASCE 4-98 (Reference 11) and the total response is obtained by superposition method. The modal superposition method may be used in dynamic analyses of seismic Category I SSCs.
Direct Integration Method

In this method, the direct integration of the equations of motion by implicit or explicit methods of numerical integration is used to solve the equations of motion. In general implicit methods, ΔT is not larger than 1/10 of the shortest period of interest. The direct integration method is used to validate coarse mesh model to be used in the seismic analysis of the nuclear island structures versus fine mesh model under the fixed-base condition.

Complex Frequency Response Method

The equation of motion can also be solved in the frequency domain using the complex frequency response method. In this method, the transfer functions are first determined and the applied forces are then transformed into the frequency domain. The fast Fourier transform (FFT) algorithm is commonly used for the transformation between the time domain and frequency domain. To facilitate the FFT operation, the total number of digitized points of the excitation time history that is used is a power of 2, which can be achieved by a process known as zero padding, which involves adding trailing zeros to the input ground motion. For damped systems, the trailing zeros also serve as a quiet zone, which allows the transient response motions to die out at the end of the duration to avoid cyclic overlapping in the discrete Fourier transform procedure.

The seismic responses of seismic Category I structures are obtained from site-independent analyses performed using three-dimensional soil-structure interaction models with the program ACS SASSI (Reference 12), which utilizes a time-history analysis in the frequency domain with substructuring techniques and complex stiffness representation of stiffness and damping properties of the structures and subgrade. The global complex stiffness matrix of the structure is assembled from the stiffness matrices of the finite elements.

With the substructuring techniques, impedance, load vector, and complex dynamic stiffness matrices are developed separately for the structure and the subgrade. The input ground motion is transformed into the frequency domain using FFT. The equations of motion for the complex soil-structure interaction system are then developed by combining the equations of motion for the structure with those of the subgrade in the frequency domain. The seismic responses are obtained in the frequency domain from solution of complex algebraic equations for a selected set of frequencies of analysis. The analysis solutions
obtained for the selected set of frequencies are then interpolated and transformed into the
time domain using inverse FFT.

3.7.2.2  Natural Frequencies and Responses

The modal analyses of the FEMs representing the seismic Category I structures are
performed separately using the ANSYS (Reference 13) or GTSTRUDL (Reference 14)
computer program. A total of 529 modes and their frequencies from the modal analysis of
the FEM for the reactor containment building are computed. Figures 3.7-28, 3.7-29, and
3.7-30 show the first major X-, Y-, and Z-mode shapes of the containment structure,
respectively, and modal frequencies and participating mass ratios are summarized in Table
3.7-10. Figures 3.7-31, 3.7-32, and 3.7-33 show the first major X-, Y-, and Z-mode shapes
of the internal structure, respectively, and modal frequencies and participating mass ratios
are summarized in Table 3.7-11.

A total of 2,500 modes and their frequencies from the modal analysis of the FEM for the
auxiliary building are computed. Figures 3.7-34, 3.7-35, and 3.7-36 show the first major
X-, Y-, and Z-mode shapes of the auxiliary building, respectively, and modal frequencies
and participating mass ratios are summarized in Table 3.7-12.

A total of 150 modes and their frequencies from the modal analysis of the FEM for the
emergency diesel generator building are computed. Figures 3.7-37, 3.7-38, and 3.7-39
show the first major X-, Y-, and Z-mode shapes, respectively, and modal frequencies and
participating mass ratios are summarized in Table 3.7-13.

The seismic response parameters resulting from the combined effect of both horizontal and
one vertical seismic input motions are obtained by using the square root of the sum of the
squares (SRSS) method. The soil-structure interaction analyses for eight soil profiles
developed to represent generic site conditions and one fixed-base analysis are performed
for the seismic Category I structures. The final analysis results are obtained by
enveloping both soil-structure interaction analysis results and fixed-base analysis results.

The seismic responses maximum absolute nodal accelerations, maximum displacements
relative to the top of foundation mat, and maximum member forces for the reactor
containment building containment structure, reactor containment building internal structure,
auxiliary building, and emergency diesel generator building are presented in Tables 3.7-14
through 3.7-25. The maximum response acceleration values are obtained from the peak of
the response time histories computed by soil-structure interaction analysis.
3.7.2.3 Procedures Used for Analytical Modeling

3.7.2.3.1 Designation of Systems versus Subsystems

The calculation of the dynamic response of a nuclear power plant subject to an earthquake loading is divided into two categories. The first is the safety-related main structural system and the second is the safety-related subsystem. The safety-related main structural system category refers to the analysis of standard plant buildings and structures that house and/or support safety-related systems. The safety-related subsystems category refers to smaller safety-related SSCs supported by the safety-related main structural systems.

The safety-related structures that are analyzed in the main structural system analysis are:

a. Reactor containment building prestressed concrete containment structure and reinforced concrete internal structure
b. Reinforced concrete auxiliary building
c. Reinforced concrete emergency diesel generator building

3.7.2.3.2 Decoupling Criteria for Subsystems

As recommended in the SRP 3.7.2 (Reference 15), the following decoupling criteria are used when a subsystem needs to be modeled:

a. If \( R_m < 0.01 \), decoupling is performed for any \( R_f \)

b. If \( 0.01 < R_m < 0.1 \), decoupling is performed if \( R_f \leq 0.8 \) or \( R_f \geq 1.25 \)

c. If \( R_m > 0.1 \), an approximate dynamic model of the subsystem is included in the main structural system

Where:

\[
R_m = \frac{\text{total mass of the supported subsystem}}{\text{total mass of the supporting system}}
\]

\[
R_f = \frac{\text{fundamental frequency of the supported subsystem}}{\text{dominant frequency of the support motion}}
\]
In general, most subsystems such as equipment and piping, with the exception of the reactor coolant system (RCS) and of the polar crane in reactor containment building, are decoupled from the floor that supports them.

3.7.2.3.3 Modeling of Safety-related Structures

Safety-related structures are modeled as three-dimensional FEMs. Major structural element systems such as floor slabs, foundation mat, roof slab, shear walls, and main frames are included in the FEM. All subsystems such as equipment and piping are considered in accordance with the decoupling criteria described in Subsection 3.7.2.3.2. For all seismic analyses, the dead load, live load, and attachment load to piping, cable tray and miscellaneous equipment mass are assumed to contribute to the inertial forces. In addition, mass associated with all heavy equipment is also included in the computation of floor or wall masses.

In general, in addition to the mass of the tributary structure dead load, additional mass equivalent to 25% of the specified floor live load or 75% of the specified snow load, where applicable; 2.394 kN/m² (50 psf), the equivalent of the miscellaneous dead load supported on the floors; and 0.479 kN/m² (10 psf), the equivalent of the attachment load on each face of the walls are included in the calculation of the inertia forces of the FEM.

Some exceptions are considered for floors in the reactor containment building and the roof of the auxiliary building. The seismic live load is not considered for floors in the seismic analysis model of the reactor containment building because the effect of seismic live load on the reactor containment building seismic response has been shown by analysis to be negligibly small. The 2.394 kN/m² (50 psf) load equivalent to the roofing material dead load is only considered for the roof of the auxiliary building due to the containment dome not having roofing material.

The reactor containment building consists of the containment structure, internal structure, and RCS. The structures in the reactor containment building share a mat foundation with the auxiliary building. The containment structure is a prestressed concrete structure and the internal structures are made of reinforced concrete. The RCS is connected to the internal structure. Appendix 3.9B provides a detailed description of the RCS structural model. Each substructure is modeled separately, and they are then combined into a total model for the seismic analysis.
3.7.2.3.3.1 Models for Nuclear Island Structures

The models for seismic excitation of the safety-related structures for the APR1400 standard plant design consist of the following structures:

a. Reactor containment building containment structure

b. Reactor containment building internal structure

c. Auxiliary building

The modeling approach used is to develop a separate three-dimensional FEM for each of the structures. The seismic analysis models of the containment structure, internal structure, and auxiliary building are shown in Figures 3.7-24, 3.7-25, and 3.7-26, respectively.

The detailed procedure for the development of the seismic analysis models regarding the nuclear island structures is provided in Technical Report APR1400-E-S-NR-14002-P (Reference 16).

FEM of Reactor Containment Building Containment Structure

The reactor containment building containment structure is a post-tensioned concrete structure consisting of a cylindrical wall and hemispherical dome with three buttresses spaced 120 degrees apart circumferentially. The reactor containment building containment structure is modeled using quadrilateral shell elements as shown in Figure 3.7-24.

FEM of Reactor Containment Building Internal Structure

For the reactor containment building internal structure, the load-resisting elements consist of reinforced concrete walls. These walls are modeled with quadrilateral shell elements or eight-node solid elements depending on the thickness of the walls. Concrete slabs are modeled with quadrilateral shell elements. The internal structure basement is modeled with eight-node solid and four-node tetrahedral solid elements. Figure 3.7-25 shows the FEM of the reactor containment building internal structure.
FEM of Auxiliary Building

The auxiliary building is composed of rectangular reinforced concrete walls and reinforced concrete slabs, which are lateral load-resisting systems and frames that support the vertical loads. The walls and slabs are modeled with quadrilateral shell elements. The frames are modeled with beam elements. Figure 3.7-26 shows the FEM of the auxiliary building.

Combined Model of Nuclear Island Structures

The dynamic model of the RCS is coupled to the internal structure FEM. The nuclear island structures share one common basemat, the reactor containment building and auxiliary building structural models are combined with each other on the common basemat to form a combined model used for the seismic analysis of the nuclear island structures.

3.7.2.3.3.2 Model for Emergency Diesel Generator Building

The emergency diesel generator building consists of a building structure to accommodate the emergency diesel generator and diesel fuel oil tank room, which is separated from the building structure. The emergency diesel generator building and diesel fuel oil tank room FEMs are individually developed following the procedure used in the development of the auxiliary building model. Figure 3.7-27 shows FEMs of the emergency diesel generator building.

3.7.2.3.4 Modeling for Three Component Input Motions

The three-dimensional FEMs of the safety-related structures as described in Subsection 3.7.2.3.3 are used in separate analyses for each of the three components of design time histories prescribed on the surface of the finished grade.

3.7.2.4 Soil-Structure Interaction

The soil-structure interaction analysis of the seismic Category I structures is performed using the substructuring method formulated in the frequency domain using the complex response method and the finite element method. For the soil-structure interaction analysis, the methodology of the ACS SASSI computer program (Reference 12) is used. ACS SASSI consists of a number of program modules used to solve dynamic soil-structure interaction problems in a seismic environment. The flexible volume method (the so-called direct method) of the ACS SASSI analysis methodology is used. For this analysis, the
seismic environment is defined by vertically propagating S-wave for the horizontal excitation and by vertically propagating P-wave for the vertical excitation.

In a substructuring method, the soil strata and half-space are assembled and condensed first for computation of transfer functions in the frequency domain. From this, the impedances at the soil-structure interface are established. Subsequently, the impedances are combined with a model of the structure, the control motion is applied to the combined system, and the equations of motion are solved for computation of final accelerations and displacements.

A detailed description of the seismic responses obtained from the soil-structure interaction analysis for the seismic Category I structures is summarized in Appendix 3.7A.

3.7.2.5 Development of In-Structure Response Spectra

The time-history analysis using complex frequency response method is used to generate the floor response spectra at wall and floor locations in the FEMs. The spectra are generated according to the procedure given in NRC RG 1.122 (Reference 17). As described in Subsection 3.7.2.3.4, the amplified response spectra in horizontal and vertical directions are obtained by three separate analyses. The response spectra resulting from the combined effect of both horizontal and one vertical seismic input motions are obtained by using the square root of the sum of the squares (SRSS) method. The effects of vertical floor flexibility are explicitly included.

The spectra are generated for appropriate critical damping values. The peaks of the response spectra are broadened as described in NRC RG 1.122 and are referred to as in-structure response spectra (ISRS).

3.7.2.6 Three Components of Earthquake Motion

For dynamic analyses of the seismic Category I structures, three statistically independent orthogonal components of earthquake motion, two horizontal and one vertical, are applied to the structural models as separate loading cases. The models are analyzed using either the time-history analysis method in time domain or frequency domain, or response spectrum analysis methods as appropriate. For the time-history analysis, the total response can be obtained by algebraically summing the response parameters. For the response spectrum analysis, the total response of the structure due to the three input seismic motions is obtained by combining the directional responses using the SRSS method.
The peak responses due to the three earthquake components from the response spectrum analysis can also be combined using 100-40-40 percent rule, as described in NRC RG 1.92 (Reference 18). The 100-40-40 percent rule is based on the observation that the maximum increase in the resultant for two orthogonal forces occurs when these forces are equal. The maximum value is 1.4 times of one component. All possible combinations of the three orthogonal responses are considered. The 100-40-40 combination is expressed mathematically as:

\[ R = (\pm 1.0RX \pm 0.4RY \pm 0.4RZ), \text{ or} \]
\[ R = (\pm 0.4RX \pm 1.0RY \pm 0.4RZ), \text{ or} \]
\[ R = (\pm 0.4RX \pm 0.4RY \pm 1.0RZ). \]

The 100-40-40 percent rule may apply to combining spatial components of responses in the same direction due to different components of motion.

3.7.2.7 Combination of Modal Responses

The total seismic response of a structure to an input response spectrum loading is obtained by combining the corresponding maximum individual modal responses of the structure in accordance with the requirements of NRC RG 1.92 (Reference 18). Beyond the amplified spectral acceleration region, the modal responses consist of both the periodic and rigid components. The periodic components of modal responses are combined using the following double sum equation:

\[ R_{pl} = \left[ \sum_{i=1}^{n} \sum_{j=1}^{n} \varepsilon_{ij} R_{pi} R_{pj} \right]^{1/2} \]

where \( R_{pl} \) is combined periodic response for the \( I_{th} \) component of seismic input motion (\( I=1, 2, \) and 3, for one vertical and two horizontal components), \( \varepsilon_{ij} \) is the modal correlation coefficient for modes \( i \) and \( j \), \( R_{pi} \) and \( R_{pj} \) are periodic components of responses of modes \( i \) and \( j \), respectively, and \( n \) is number of modes considered.

If the frequencies of the modes are sufficiently separated, double sum equation is equivalent to the SRSS method. If modes with closely spaced frequencies exist and are defined according to the critical damping ratio, the modal correlation coefficient should be
determined appropriately considering modal frequencies, modal damping ratios, and the time duration of earthquake motion using the SRSS method from NRC RG 1.92.

The rigid response component of a modal response, $R_i$, is defined as follows:

$$ R_i = \alpha_i R_i $$

The periodic response component of $R_{pi}$ can then be expressed as follows:

$$ R_{pi} = \left[1 - \alpha_i^2\right]^{1/2} R_i, \quad \text{where } R_i^2 = R_{pi}^2 + R_{ri}^2 $$

The rigid response coefficient, $\alpha_i$, can be obtained from the equations in NRC RG 1.92. The rigid responses are combined algebraically, as follows:

$$ R_{ri} = \sum_{i=1}^{n} R_{ri} + R_{\text{missmass}i} $$

where $R_{\text{missmass}i}$ is the residual rigid response of the missing mass modes for the $i_{th}$ component of seismic input motion. The effect of missing mass modes not included in the analysis is accounted for by using the method given in NRC RG 1.92.

Finally, the combined response is calculated as follows:

$$ R_i = \left[ R_{ri}^2 + R_{pi}^2 \right]^{1/2} $$

3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

The interfaces between seismic Category I and non-seismic Category I structures are designed for the dynamic loads and displacements produced by both the seismic Category I and non-seismic Category I structures.

To provide reasonable assurance that the failure of a non-seismic Category I structure under the effect of a seismic event does not impair the integrity of an adjacent seismic Category I structure, one of the following criteria is used:

a. Maintenance of sufficient separation between non-seismic Category I structures and seismic Category I structures
b. Analysis and design of non-seismic Category I structures to prevent their failure under SSE conditions

c. Design of seismic Category I structures to withstand loads due to the collapse of the adjacent non-seismic Category I structures if sufficient spatial separation is not achieved

The seismic Category II turbine generator building and compound building are analyzed and designed to prevent their failure under SSE conditions (criterion b). Since the seismic Category II alternate alternating current gas turbine generator building is located at a considerable distance from the seismic Category I structures, as shown in Figure 1.2-1, criterion a applies to the alternate alternating current gas turbine generator building.

The turbine generator building and compound building are located on the west side and south side of nuclear island with a 900 mm (3 ft) gap on each side. Figures 3.7-40 and 3.7-41 show the FEMs of the turbine generator building and compound building, respectively. To evaluate the structure-soil-structure interaction (SSSI) effects on the nuclear island structures due to presence of adjacent non-seismic Category I structures, the SSSI analysis using the coupled model for entire structures is performed. The interaction effects of these non-seismic Category I structures on the nuclear island are negligible as provided in Technical Report, APR1400-E-S-NR-14005-P (Reference 19).

The COL applicant is to confirm that any site-specific non-seismic Category I structures are designed not to degrade the function of a seismic Category I SSC to an unacceptable safety level due to their structural failure or interaction. The relative displacements calculated by the COL applicant are not to exceed the gaps between seismic Category I and non-seismic Category I structures.

For the seismic analysis of the seismic Category II structures, the site-specific FIRS are applied as seismic input motions and a site-specific soil profile is to be established as a supporting media. The same seismic analysis procedure which is applied to the seismic Category I structures is also applied to the seismic Category II structures.

The design codes used by the COL applicant for the structural design of the seismic Category II structures are as follows:

a. The design of the turbine generator building is performed using ACI 318 or AISC 360 as described in Table 3.2-1.
b. The compound building is designed according to RW-IIa criteria in NRC RG 1.143. Hence, the design of the compound is performed using ACI 349 or AISC N690 as described in Table 3.2-1.

c. The design of the alternate alternating current gas turbine generator building located on the southwest side of the plant is performed using ACI 318 as described in Table 3.2-1.

The potential effects of sliding and uplift for the seismic Category II structures are checked using the same approach applied in the stability check for the seismic Category I structures. The COL applicant is to evaluate the pressures on the below grade walls of the NI, resulting from site-specific SSSI effects between the NI and adjacent seismic Category II structures (COL 3.7(4)).

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

To consider variations in the structural frequencies due to the uncertainties in material properties of the structure and approximations in modeling, the peaks of the computed floor response spectra are broadened by ± 15 percent and smoothed in accordance with NRC RG 1.122, as described in Subsection 3.7.2.5.

The effects of potential concrete cracking on the structural stiffness of reinforced concrete structures are considered as enveloping the floor response spectra for cracked concrete properties with 7 percent damping for the reinforced concrete structures and those for uncracked properties with 4 percent damping for the reinforced concrete structures.

Both uncracked and cracked concrete stiffnesses are considered separately in the seismic analysis models of the seismic Category I structures. For consideration of potential concrete cracking, the cracked concrete stiffness in horizontal and vertical seismic analysis models is reduced by half of the uncracked concrete stiffness except prestressed concrete containment structure and reinforced concrete columns and walls in the vertical models described in ASCE/SEI 43-05 (Reference 20). Therefore, for eight soil profiles and one fixed-base condition, a total of 18 analysis cases are performed in the seismic analysis to generate the floor response spectra of the seismic Category I structures.

The selected locations where the floor response spectra are obtained in analysis models and resultant floor response spectra enveloping the 18 analysis cases for the nuclear island structures are provided in Technical Report, APR1400-E-S-NR-14003-P (Reference 21).
3.7.2.10 Use of Constant Vertical Static Factors

The safety-related main structural systems are analyzed in the vertical direction using the methods described in Subsection 3.7.2.1. The vertical component is considered to occur simultaneously with the two horizontal components and consistently combined with the horizontal components of the seismic motion as described in Subsection 3.7.2.6. Therefore, a constant vertical static factor is not used for the seismic design of seismic Category I structures.

3.7.2.11 Methods Used to Account for Torsional Effects

Because the structural models used for seismic Category I structures are constructed with finite elements containing 6 degrees of freedom per node, incorporating torsional effects into the models, the mathematical models include sufficient mass points and corresponding dynamic degrees of freedom to provide a three-dimensional representation of the dynamic characteristics of the structure.

Torsional effects are also accounted for in the structural models used to generate floor response spectra. An additional eccentricity of 5 percent of the maximum building dimension, perpendicular to load direction that results in an accidental torque, is applied to the static finite element structural model to calculate element forces due to accidental torsion. Accidental torsion is considered in both the E-W and N-S directions.

3.7.2.12 Comparison of Responses

The time-history analysis method based on complex frequency response method is used for the seismic analysis of seismic Category I structures. The response spectrum analysis is used to compute the seismic design forces of the containment structure and internal structure in the reactor containment building using the in-structure response spectra at the top of the basemat generated from the seismic soil-structure interaction analysis. The responses from these two methods are compared and provided in Technical Report, APR1400-E-S-NR-14003-P (Reference 21).

3.7.2.13 Methods for Seismic Analysis of Dams

The COL applicant is to perform seismic analysis for any site-specific seismic Category I dams, if necessary (COL 3.7(5)).
3.7.2.14 **Determination of Dynamic Stability of Seismic Category I Structures**

The design overturning moments and base shears for seismic Category I structures are determined by time-history analysis based on the complex frequency response method. The seismic motion is input separately to the structural models in three independent orthogonal directions. To check the overturning and sliding, the simultaneous action of horizontal and vertical seismic forces using methods described in Subsection 3.7.2.6 is incorporated.

The procedure to check the stability of seismic Category I structures is described in Subsection 3.8.5.

3.7.2.15 **Analysis Procedure for Damping**

For modal superposition method, composite modal damping values are used for structures with components of different damping characteristics. The composite modal damping values are based on weighting the damping factors according to the mass or the stiffness of each element. For the mass-weighted damping, the formulation is as follows:

\[
\beta_j = \frac{\sum_{i=1}^{N} \{\phi_i\}^T \beta_i \{M_i\} \{\phi_j\}}{\{\phi_j\}^T [M] \{\phi_j\}}
\]

Where:

- \(\beta_j\) = composite modal damping for mode \(j\)
- \(N\) = total number of components
- \(\beta_i\) = critical modal damping associated with component \(i\)
- \(\phi_j\) = mode shape vector
- \(\{M_i\}\) = subregion of mass matrix associated with component \(i\)
- \([M]\) = mass matrix of the system
For the stiffness-weighted damping, the formulation is as follows:

\[
\beta_j = \sum_{i=1}^{N} \phi_j^T \beta_i \{K_i\} \phi_j \\
\sum_{i=1}^{N} \phi_j^T [K] \phi_j
\]

Where:

\[
\{K_i\} = \text{subregion of stiffness matrix associated with component } i \\
[K] = \text{stiffness matrix of the system}
\]

For direct integration method, viscous damping proportional to the mass and stiffness matrix is used; thus

\[
[C] = \alpha [M] + \beta [K]
\]

where [C] is the damping matrix, [K] is the stiffness matrix, and [M] is the mass matrix. The values of \(\alpha\) and \(\beta\) are selected so that the damping in the range of frequency of interest is approximately equal to the damping of the structure.

3.7.3 Seismic Subsystem Analysis

This subsection describes the seismic analysis methods for the APR1400 seismic Category I subsystems as civil structures that are not included in the main structural system, such as miscellaneous concrete and steel structures, buried conduit, tunnel, dam, and above-ground tanks.

The seismic analysis of the seismic Category I mechanical subsystems, such as piping and equipment, is described in Section 3.9.

3.7.3.1 Seismic Analysis Methods

The seismic analysis of seismic Category I subsystems is performed using either the response spectrum analysis or time-history analysis, as described in Subsection 3.7.2.1, or the equivalent static method described in Subsection 3.7.3.1.1.
3.7.3.1.1  Use of Equivalent Static Load Method of Analysis

In the seismic analyses of components, the equivalent static load method would be used if a
dynamic analysis is not performed. When the equivalent static load method is used for the
seismic analysis of components, i) justification of the adequacy of the analysis models and
conservatism of the analysis results are to be provided by showing that the analysis results
obtained from the equivalent static load method are more conservative than those of a
dynamic analysis, such as response spectrum analysis method or time history analysis
method, ii) the responses obtained from relative motion between points of support, if any,
are combined with the response from the inertial loads, and iii) the constant static factor is
equal to 1.5 times the peak spectral acceleration in the applicable required response spectra.
A value less than 1.5 times could be used if its conservatism and justification are verified.
The equivalent seismic static load is the product of the equipment or component mass and
the constant static factor.

3.7.3.1.2  Determination of Number of Earthquake Cycles

The procedure used to account for the fatigue effect of cyclic motion associated with
seismic excitation recognizes that the actual motion experienced during a seismic event
consists of a single maximum or peak motion and some number of cycles of lesser
magnitude. The total or cumulative usage factor can also be specified in terms of a finite
number of cycles of the maximum or peak motion. Based on this consideration, seismic
Category I subsystems, components, and equipment are designed for a total of two SSE
events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress
range). Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full
SSE vibratory cycles may be used (but with an amplitude not less than one-third \([1/3]\) of
the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Std.
344 (Reference 22).

3.7.3.2  Procedure Used for Analytical Modeling

The criteria and bases described in Subsection 3.7.2.3 are used to determine whether a
component or structure will be analyzed as a subsystem. The modeling techniques
incorporate either a single- or multi-degree of freedom subsystem consisting of discrete
masses connected by spring elements. The associated damping coefficients are consistent
with Table 3.7-7. The degree of complexity of each model is sufficient to accurately
evaluate the dynamic behavior of the component.
3.7.3.3 Analysis Procedures for Damping

The analysis procedure used to account for the damping in subsystems conforms with Subsections 3.7.1.2 and 3.7.2.15.

3.7.3.4 Three Components of Earthquake Motion

Seismic responses resulting from analysis of subsystems due to three components of earthquake motions are combined in the same manner as the seismic response resulting from the analysis of building structures as specified in Subsection 3.7.2.6.

3.7.3.5 Combination of Modal Responses

For combination of modal responses of subsystems, Subsection 3.7.2.7 should be referred for details. When a response spectrum method of analysis is used to analyze a subsystem, the maximum responses such as accelerations, shears, and moments in each mode are calculated regardless of time. If the frequencies of the modes are well separated, the SRSS method of mode combination gives acceptable results; however, where the structural frequencies are not well separated, the modes are combined in accordance with NRC RG 1.92. The criteria for determining how modal frequencies are separated are specified in NRC RG 1.92, as the definition of modes with closely spaced frequencies. The definition of modes with closely spaced frequencies is a function of the critical damping ratio and is as follows;

a. For critical damping ratios \( \leq 2\% \), modes are considered closely spaced if the frequencies are within 10% of each other (i.e., for \( f_i < f_j \), \( f_j \leq 1.1 f_i \))

b. For critical damping ratio > 2%, modes are considered closely spaced if the frequencies are within five times the critical damping ratio of each other (i.e., for \( f_i < f_j \) and 5% damping, \( f_j \leq 1.25 f_i \), for \( f_i < f_j \) and 10% damping, \( f_j \leq 1.5 f_i \))

3.7.3.6 Use of Constant Vertical Static Factors

The seismic analysis of seismic Category I subsystems can be performed using the response spectrum analysis method, the time history analysis method, or the equivalent static load method, as specified in Subsection 3.7.3.1. When the equivalent static load method is employed, the constant vertical static factor can be used to calculate the vertical response loads for the seismic design of seismic Category I subsystems only if it can be
demonstrated that the subsystems are rigid in the vertical direction or the acceptance criteria in SRP 3.7.2, II.1.B are satisfied.

### 3.7.3.7 Buried Seismic Category I Conduits and Tunnels

During an earthquake, buried structures such as conduits and tunnels respond to various seismic waves propagating through the surrounding soil as well as to the dynamic differential movements of the buildings to which the structures are connected. The various waves associated with earthquake motion are P (compression) waves, S (shear) waves, and Rayleigh waves. The stresses in the buried structure are governed by the velocity and angle of incidence of these traveling waves. However, the wave types and their directions during an earthquake are very complex. For design purposes, the seismic-induced upper-bound strains and corresponding stresses in the buried concrete electrical ducts are calculated using expressions given by ASCE 4-98 (Reference 11).

Seismic design for buried seismic Category I structures takes the effect of wave propagation into consideration, based on the assumption that there is no movement of the buried structure remote from anchor points relative to the surrounding soil referred to in ASCE 4-98, Subsection 3.5.2. That is, the strain of the structure is the same as that of the surrounding soil medium, and the stress of the structure is calculated from the strain. Consideration of relative deformation between anchor points and the adjacent soil is applied to the design using the SRSS method for the three orthogonal stresses calculated from the relative displacements of the seismic analysis results.

The resistance effect of the surrounding soil for deformation or displacement of the buried structures, differential movement of the anchors, and shape or curvature changes of the bent parts is taken into account in the analysis. The structures can be modeled by beam elements supported by an elastic foundation representing the stiffness of the adjacent soil.

Lateral dynamic soil pressure on buried seismic Category I structures is calculated in accordance with elastic theory by Wood referred to in ASCE 4-98, Subsection 3.5.3. The effect of underground water is considered by applying the equation proposed by Matuo and O’Hara based on the theory from Westergaard that is referred to in ASCE 4-98, Subsection 3.5.3.1.

The COL applicant is to perform seismic analysis of buried seismic Category I conduits and tunnels (COL 3.7(6)).
3.7.3.8 Methods for Seismic Analysis of Category I Concrete Dams

The COL applicant is to perform seismic analysis for any site-specific seismic Category I dams, if necessary (COL 3.7(5)).

3.7.3.9 Methods for Seismic Analysis of Above-ground Tanks

Above-ground seismic Category I tanks are large, flat-bottomed, single-shell, cylindrical tanks anchored to reinforced concrete pads. Seismic analysis procedures address the issues described in NUREG/CR-1161RD (Reference 23), pages 28-30, based primarily on the methods of Haroun and Housner (Reference 24). The hydrodynamic mass effects following the procedures described in ASCE 4-98, Subsection 3.1.6, are considered in the seismic analysis model.

Because of the symmetry of these vertical tanks, the larger of the two horizontal earthquake components, if they are not equal in magnitude, is combined by the SRSS method with the vertical earthquake component.

The assessment of dynamic loading on storage tanks verifies stability of the tank wall against buckling behavior, accounting for hydrodynamic loads (impulsive and convective) and shell flexibility.

In the generation of dynamic loads, tanks are evaluated as filled, with consideration of convective (sloshing), impulsive (fluid-shell interaction), and rigid modes of behavior. For the convective mode, fluid damping is taken as 0.5 percent of critical damping in accordance with NRC RG 1.61. For the impulsive mode, structural (tank wall) damping is taken for the SSE, in accordance with SRP 3.7.3 (Reference 25). The effective mass, its location, and natural frequency for each mode of behavior are obtained from the equations and graphs in Haroun and Housner.

Using the site-specific foundation input response spectra developed at the base of the tank, spectral accelerations obtained for each mode at the appropriate damping and frequency are applied to the computation of appropriate effective mass.

Structural adequacy of the anchorage provisions for the tank (e.g., anchor bolts, embedments) is developed assuming that the overturning moment on the tank is resisted only by compression in the shell and tension in the anchor bolts. The overturning moment
at the base of the tank is computed as the sum of the flexible and rigid mode responses, each of which is the product of the applicable mass, height, and spectral acceleration.

For the seismic Category I tanks constructed as a part of buildings, the seismic analyses are performed considering input motions in three directions, two horizontal and one vertical, as the building is analyzed according to Subsection 3.7.2.3.

For seismic Category I tanks installed in buildings, the seismic analyses are performed in accordance with Subsection 3.9.2.2.14.

The COL applicant is to perform seismic analysis for the seismic Category I above-ground tanks (COL 3.7(7)).

3.7.3.10 Basis for Selection of Frequencies

The fundamental frequencies of components and equipment are selected to be less than one-half or more than twice the dominant frequencies of the support structure to avoid resonance. The equipment must be adequately designed for the applicable loads if the equipment frequencies are within this range.

3.7.3.11 Interaction of Other Systems with Seismic Category I Systems

The non-seismic Category I subsystems are designed to be isolated by either a constraint or barrier or are remotely located from any seismic Category I SSC. Otherwise, adjacent non-seismic Category I subsystems are analyzed according to the same seismic criteria as applicable to seismic Category I SSC. For non-seismic Category I subsystems attached to seismic Category I SSCs, the dynamic effects of the non-seismic Category I subsystems should be simulated in the modeling of the seismic Category I SSC. The attached non-seismic Category I subsystems, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the seismic Category I SSC.

3.7.3.12 Multiply-Supported Equipment and Components with Distinct Inputs

The seismic response of multiply-supported equipment and components with distinct inputs are obtained by either the response spectrum approach or the time history approach.
Of the response spectrum approaches, the uniform support motion (USM) method is applied with a uniform response spectrum (URS) that envelops all of the individual response spectra at the various support locations. In addition, the maximum relative support displacements are imposed on the supports in the most unfavorable combination. The final responses are obtained by the absolute summation of responses due to inertial effects and relative displacements. As an alternative to the USM method, the independent support motion (ISM) method can be employed such that all of the criteria presented in NUREG-1061 related to the ISM method are satisfied.

When the time history approach is applied, time histories of support motions may be used as input excitations to the subsystems.

### 3.7.3.13 Torsional Effects of Eccentric Masses

To consider the torsional effects of eccentric masses in seismic Category I subsystems, the eccentric masses are included in the mathematical model as eccentric masses located in their center of gravity coupled by, as applicable, either rigid members or elastic members with their own properties.

### 3.7.4 Seismic Instrumentation

Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plant,” of 10 CFR Part 50 (Reference 2), requires that suitable instrumentation be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake. It also requires shutdown of the nuclear power plant if vibratory ground motion exceeds the OBE ground motion.

The seismic monitoring system meets the relevant requirements of 10 CFR Part 50, Appendix S. The seismic monitoring system provides information on the vibratory ground motion and the resultant vibratory responses of the representative seismic Category I structures during a seismic event.

The seismic monitoring system is not a safety system and does not have any effect on safety systems or equipment. The seismic monitoring system is designed for high accuracy, reliability, and to minimize the maintenance and surveillance activities required to support the system.
3.7.4.1 Comparison with NRC Regulatory Guide 1.12

The seismic monitoring system is designed in accordance with NRC RG 1.12 (Reference 26). The instrument type, location, operability, installation, activation, and surveillance requirements are described in the Subsections 3.7.4.2 through 3.7.4.5.

The time-history accelerographs installed at appropriate locations, which are described in NRC RG 1.12, provide time-history data on the seismic responses of the free-field, containment structure, and other seismic Category I structures.

3.7.4.2 Location and Description of Instrumentation

The seismic instruments are located so that their respective responses can be compared and evaluated with appropriate design bases so that occupational radiation exposures to the plant personnel can be kept as low as reasonably achievable (ALARA) when operating and maintaining the seismic instruments. Instruments that enable the prompt processing of data at the plant site are used. The seismic monitoring system consists of time-history accelerographs, including accelerometers, accelerometer-based seismic triggers, recorders, and a seismic monitoring system cabinet.

All components of the seismic monitoring system are qualified to the provisions of IEEE Std. 344 (Reference 23).

The instruments are not mounted directly on plant equipment, piping, or supports since data obtained at these locations could be obscured by the vibratory motion associated with normal plant operation. The components are located in areas that are not generally affected by plant operational hazards such as explosions, fire, and internal flooding.

The in-structure instruments are placed at mass-point locations based on the building dynamic model so that the measured motion can be directly compared with the design spectra.

Provisions are made for instruments that have a sensor located in an inaccessible area to have data recording in an accessible location, and the instrument provides an external remote alarm to indicate a seismic event.

The COL applicant that references the APR1400 design certification will determine whether essentially the same seismic response from a given earthquake is expected at each unit in a multi-unit site or each unit is to be provided with a separate set of seismic
instruments (COL 3.7(8)). In the event that only one unit is instrumented, annunciation is provided to each control room.

A triaxial time-history accelerograph is provided at the following locations:

a. Free-field

b. Containment structure (one at the foundation and two at higher elevations)

c. Independent seismic Category I structure (one at the foundation and one at a higher elevation)

The free-field sensor is located at a place where effects of surface features, buildings, and equipment are minimal and insignificant for recording the ground motion.

The COL applicant is to confirm details of the locations of the triaxial time-history accelerographs (COL 3.7(9)).

3.7.4.2.1 Field-Mounted Sensors

The free-field instrument data are used to determine whether the OBE threshold has been exceed. Foundation-level instruments provide data on the actual seismic input to the containment structure and other seismic Category I structures, and are used to quantify the difference between the vibratory ground motion at the free-field and the building structure foundation level. The instruments have a triaxial design and one of the orthogonal axes is vertical. The remaining two are on the perpendicular plane to the vertical and are perpendicular to each other.

3.7.4.2.2 Accelerometers

The accelerometers have a triaxial design with three orthogonal axes, similar to the field-mounted sensors described above. The dynamic range of the device has a zero-to-peak ratio of at least 1,000:1 (e.g., 0.001 g to 1.0 g). The frequency range is at least 0.20 Hz to 50 Hz. The triaxial accelerometer measures acceleration as a function of time in three mutually perpendicular directions and transmits this signal to recorders at the seismic monitoring system cabinet after any one of seismic triggers is actuated. The accelerometer-based seismic triggers activate the recorders whenever the seismic trigger threshold is exceeded for any one of the three axes. The accelerometer-based seismic triggers are located in the seismic monitoring system cabinet.
3.7.4.2.3 Seismic Monitoring System Cabinet

The seismic monitoring system cabinet is provided in the instrumentation and control (I&C) equipment room and includes the following as a minimum:

a. Seismic monitoring instrument

b. Seismic recorder(s) and playback system, annunciators, and power supply unit

The seismic monitoring system cabinet includes provisions for securely bolting and/or welding the cabinet to the floor.

3.7.4.2.4 Seismic Recorder(s) and Playback System

The sampling rate is at least 200 samples per second in each of the three directions. The dynamic range is at least 1000:1 for combined record and playback. The recording medium is capable of storage and playback without deterioration of playback signal quality. The recorder includes the accelerometer-based trigger, which has a software-based on the digital threshold algorithm and activates the archival recorder whenever the adjustable threshold is exceeded in any one of the three axes.

The accelerometer-based trigger has an actuation level between 0.001 g and 0.02 g.

The playback system, including a response spectrum analyzer, remains functional at all times to provide automatic retrieval of acceleration and evaluation of the OBE condition immediately following an event, using the response spectrum check and cumulative absolute velocity (CAV).

The instrument records, at a minimum, 3 seconds of low-amplitude motion prior to seismic trigger actuation, continues to record the motion during the period in which the earthquake motion exceeds the seismic trigger threshold, and continues to record low-amplitude motion for a minimum of 5 seconds beyond the last exceedance of the seismic trigger threshold. Additional pre-event memory (for up to 30 seconds of pre-earthquake recording) is provided for P-wave correlation.

All recording media and printers are capable of operating for a minimum of 25 minutes on backup battery power.

The playback system is powered from the system power distribution unit.
3.7.4.2.5 Power Supply

An instrument power source has sufficient capacity for a time period longer than the maintenance interval to provide required continuous standby power and sufficient power for a minimum 25 minutes of system operation at any time.

The battery has sufficient capacity to power the instrument to sense and record 25 minutes of motion over a period of not less than the channel check test interval. This can be accomplished by providing enough battery capacity for a minimum of 25 minutes in a 24-hour period without recharging, in combination with a battery charger whose line power is connected to an uninterruptible power supply (UPS).

3.7.4.3 Main Control Room Operator Notification

Triggering of the free-field or any foundation-level time-history accelerograph is annunciated in the main control room (MCR) and remote shutdown room (RSR). The operator receives audible and visual annunciation in the MCR and RSR by the triggered seismic instrument. The annunciation is provided to the operator when a time-history accelerograph is activated. Also, the annunciation is provided if one or more seismic trigger setpoints of OBE or safe shutdown earthquake (SSE) are being exceeded. Annunciation of triggering of the time-history accelerograph in the MCR, RSR, and the recorder readouts in the I&C equipment room aid the plant operator in making a decision for plant shutdown.

3.7.4.4 Comparison with NRC Regulatory Guide 1.166

When an earthquake occurs, ground motion data are recorded by the seismic instrument. These data are used to make a rapid determination of the degree of severity of the seismic event. The data from the plant’s free-field seismic instrument, coupled with information obtained from a plant walkdown, are used to make the initial determination whether the plant must be shut down.

The evaluation to determine whether the OBE was exceeded is performed using data obtained from the three orthogonal components of the free-field ground motion. The evaluation consists of a check of the response spectrum and CAV and a check on the operability of the instrument.
Exceedance of the OBE response spectrum is determined in accordance with NRC RG 1.166 (Reference 27). The CAV is calculated in accordance with NRC RG 1.166 and EPRI TR-100082 (Reference 29). The CAV limit is determined to be exceeded if the CAV calculation is greater than 0.16 g-second.

If the response spectrum and the CAV cannot be obtained because the seismic instrument or data processing is inoperable, the criteria of NRC RG 1.166 Appendix A are used to determine whether the OBE has been exceeded.

The seismic instrumentation program is designed in accordance with the guidelines of NRC RG 1.166 and EPRI NP-6695 (Reference 28).

The plan for post-earthquake walkdown inspections is established using preselected equipment and structures and is used to quantify the damage caused by the earthquake. It is also used to establish the extent of inspections, tests, and evaluations necessary to demonstrate readiness for plant restart.

The procedures for actions immediately after an earthquake contain a check of the neutron flux monitoring sensors and a check of containment isolation valves. The earthquake-induced vibration of the vessel could lead to a change in neutron flux, so a prompt check of the neutron flux monitoring sensors would provide an indication that the reactor is stable. The containment isolation valve may have malfunctioned during the earthquake. Inspection of the containment isolation system is necessary to ensure continued containment integrity.

3.7.4.5 Instrument Surveillance

The instrumentation is selected to require minimal maintenance and inservice inspection, as well as minimal time and numbers of personnel to conduct installation and maintenance. The seismic instrumentation operates during all modes of plant operation, including periods of plant shutdown. The maintenance and repair procedures will provide for keeping the maximum number of instruments in service during plant operation and shutdown.

As required by NRC RG 1.12, the seismic monitoring system is to be given channel checks every 2 weeks for the first 3 months of service after startup. After the initial 3-month period and three consecutive successful checks, monthly channel checks are sufficient. The monthly channel check is to include checking the batteries. The functional test is
performed every 6 months. The channel calibration is performed during each refueling outage at a minimum.

3.7.4.6 Program Implementation

The COL applicant is to identify the implementation milestones for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5 (COL 3.7(10)).

The milestones covered in program implementation are summarized as follows:

a. A file containing information on all the seismic instrumentation will support keeping the plant safe in conformance with NRC RG 1.166. The file includes information on each instrument type such as model, list of special features, and operation and maintenance manuals.

b. Operational procedures contain an alarm response procedure, pre-shutdown inspection procedure, and post-event inspection procedure.

1) The alarm response procedure supports immediate actions in the event of a seismic warning alarm. The alarms provide additional information on the status and performance of components and systems.

2) The pre-shutdown inspection procedure supports determination of the effects of the earthquake on essential safe shutdown equipment. Following the earthquake, the equipment must be inspected for any needed resets or repairs, as well as for readiness prior to initiating shutdown activities.

3) The post-event inspection procedure supports determination of the degree of damage to equipment and equipment acceptability for continued operation.

The COL applicant is to prepare a procedure for the post-shutdown inspection and plant restart in accordance with the guidance of NRC RG 1.167 (Reference 30) (COL 3.7(11)).

3.7.5 Combined License Information

COL 3.7(1) The COL applicant is to demonstrate the applicability of soil degradation models used in site-specific site response analysis for the site conditions.
COL 3.7(2) The COL applicant is to compare the site-specific strain-compatible soil properties with generic soil properties in order to confirm that the site meets the generic soil profile used in the standard design.

COL 3.7(3) The COL applicant is to provide the seismic design of the seismic Category I SSCs and seismic Category II structures that are not part of the APR1400 standard plant design. The seismic Category I and II structures are as follows:

a. Seismic Category I essential service water building
b. Seismic Category I component cooling water heat exchanger building
c. Seismic Category II turbine generator building
d. Seismic Category II compound building
e. Seismic Category II alternate alternating current gas turbine generator building

COL 3.7(4) The COL applicant is to confirm that any site-specific non-seismic Category I structures are designed not to degrade the function of a seismic Category I SSC to an unacceptable safety level due to their structural failure or interaction. The COL applicant is to confirm that the calculated relative displacements do not exceed the gaps between seismic Category I and non-seismic Category I structures.

The COL applicant is to apply the site-specific FIRS as seismic input motions and to establish a site-specific soil profile as a supporting media for the seismic analysis of the seismic Category II structures. The COL applicant is to apply the same seismic analysis procedure as the seismic Category I structures to the seismic Category II structures. The COL applicant is to perform the structural design of the seismic Category II structures using the design codes described in Subsection 3.7.2.8 and Table 3.2-1. The COL applicant is to check the potential effects of sliding and uplift for the seismic Category II structures using the same approach applied in the stability check for the seismic Category I structures. The COL applicant is to evaluate the pressures on the below grade walls of the
NI, resulting from site-specific SSSI effects between the NI and adjacent seismic Category II structures.

COL 3.7(5) The COL applicant is to perform seismic analysis for any site-specific seismic Category I dams, if necessary.

COL 3.7(6) The COL applicant is to perform seismic analysis of buried seismic Category I conduits and tunnels.

COL 3.7(7) The COL applicant is to perform seismic analysis for the seismic Category I above-ground tanks.

COL 3.7(8) The COL applicant that references the APR1400 design certification will determine whether essentially the same seismic response from a given earthquake is expected at each unit in a multi-unit site or each unit is to be provided with a separate set of seismic instruments.

COL 3.7(9) The COL applicant is to confirm details of the locations of the triaxial time-history accelerographs.

COL 3.7(10) The COL applicant is to identify the implementation milestones for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5.

COL 3.7(11) The COL applicant is to prepare a procedure for the post shutdown inspection and plant restart in accordance with the guidance of NRC RG 1.167.

3.7.6 References


Table 3.7-1

Spectral Amplitude of CSDRS for Control Points

<table>
<thead>
<tr>
<th>Horizontal Damping Ratio (%)</th>
<th>0.1 Hz</th>
<th>0.2 Hz</th>
<th>0.25 Hz</th>
<th>2.5 Hz</th>
<th>9 Hz</th>
<th>25 Hz</th>
<th>50 Hz</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>0.0276</td>
<td>0.111</td>
<td>0.171</td>
<td>1.275</td>
<td>1.062</td>
<td>0.511</td>
<td>0.300</td>
</tr>
<tr>
<td>3</td>
<td>0.0254</td>
<td>0.102</td>
<td>0.159</td>
<td>1.125</td>
<td>0.939</td>
<td>0.498</td>
<td>0.300</td>
</tr>
<tr>
<td>4</td>
<td>0.0238</td>
<td>0.096</td>
<td>0.147</td>
<td>1.020</td>
<td>0.852</td>
<td>0.487</td>
<td>0.300</td>
</tr>
<tr>
<td>5</td>
<td>0.0226</td>
<td>0.090</td>
<td>0.141</td>
<td>0.939</td>
<td>0.783</td>
<td>0.479</td>
<td>0.300</td>
</tr>
<tr>
<td>7</td>
<td>0.0207</td>
<td>0.084</td>
<td>0.129</td>
<td>0.816</td>
<td>0.681</td>
<td>0.464</td>
<td>0.300</td>
</tr>
<tr>
<td>10</td>
<td>0.0188</td>
<td>0.075</td>
<td>0.117</td>
<td>0.684</td>
<td>0.570</td>
<td>0.447</td>
<td>0.300</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Vertical Damping Ratio (%)</th>
<th>0.1 Hz</th>
<th>0.2 Hz</th>
<th>0.25 Hz</th>
<th>3.5 Hz</th>
<th>9 Hz</th>
<th>25 Hz</th>
<th>50 Hz</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>0.0184</td>
<td>0.075</td>
<td>0.114</td>
<td>1.215</td>
<td>1.062</td>
<td>0.511</td>
<td>0.300</td>
</tr>
<tr>
<td>3</td>
<td>0.0170</td>
<td>0.069</td>
<td>0.105</td>
<td>1.074</td>
<td>0.939</td>
<td>0.498</td>
<td>0.300</td>
</tr>
<tr>
<td>4</td>
<td>0.0159</td>
<td>0.063</td>
<td>0.099</td>
<td>0.972</td>
<td>0.852</td>
<td>0.487</td>
<td>0.300</td>
</tr>
<tr>
<td>5</td>
<td>0.0151</td>
<td>0.060</td>
<td>0.093</td>
<td>0.894</td>
<td>0.783</td>
<td>0.479</td>
<td>0.300</td>
</tr>
<tr>
<td>7</td>
<td>0.0138</td>
<td>0.057</td>
<td>0.087</td>
<td>0.777</td>
<td>0.681</td>
<td>0.464</td>
<td>0.300</td>
</tr>
<tr>
<td>10</td>
<td>0.0125</td>
<td>0.051</td>
<td>0.078</td>
<td>0.651</td>
<td>0.570</td>
<td>0.447</td>
<td>0.300</td>
</tr>
</tbody>
</table>
### Numerical Information of CSDRS

**a) Strong motion duration for all three directions (sec)**

<table>
<thead>
<tr>
<th></th>
<th>H1</th>
<th>H2</th>
<th>VT</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>9.2</td>
<td>10.3</td>
<td>8.4</td>
</tr>
</tbody>
</table>

**b) Peak values (A, V and D)**

<table>
<thead>
<tr>
<th></th>
<th>A (g)</th>
<th>V (in/sec)</th>
<th>D (in)</th>
</tr>
</thead>
<tbody>
<tr>
<td>H1</td>
<td>0.300</td>
<td>17.5</td>
<td>16.5</td>
</tr>
<tr>
<td>H2</td>
<td>0.300</td>
<td>17.9</td>
<td>16.1</td>
</tr>
<tr>
<td>VT</td>
<td>0.300</td>
<td>18.9</td>
<td>13.0</td>
</tr>
</tbody>
</table>

**c) Number of points of the response spectra below CSDRS**

<table>
<thead>
<tr>
<th></th>
<th>H1</th>
<th>H2</th>
<th>VT</th>
</tr>
</thead>
<tbody>
<tr>
<td>2%</td>
<td>4</td>
<td>4</td>
<td>2</td>
</tr>
<tr>
<td>3%</td>
<td>1</td>
<td>3</td>
<td>2</td>
</tr>
<tr>
<td>4%</td>
<td>4</td>
<td>3</td>
<td>2</td>
</tr>
<tr>
<td>5%</td>
<td>5</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td>7%</td>
<td>4</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td>10%</td>
<td>1</td>
<td>2</td>
<td>1</td>
</tr>
</tbody>
</table>

**d) The lowest percentage below CSDRS (%)**

<table>
<thead>
<tr>
<th></th>
<th>H1</th>
<th>H2</th>
<th>VT</th>
</tr>
</thead>
<tbody>
<tr>
<td>2%</td>
<td>-6.66</td>
<td>-4.53</td>
<td>-0.48</td>
</tr>
<tr>
<td>3%</td>
<td>-4.72</td>
<td>-1.69</td>
<td>-2.99</td>
</tr>
<tr>
<td>4%</td>
<td>-4.58</td>
<td>-5.89</td>
<td>-2.94</td>
</tr>
<tr>
<td>5%</td>
<td>-3.61</td>
<td>-4.15</td>
<td>-1.00</td>
</tr>
<tr>
<td>7%</td>
<td>-6.11</td>
<td>-5.51</td>
<td>-0.62</td>
</tr>
<tr>
<td>10%</td>
<td>-6.02</td>
<td>-2.40</td>
<td>-1.43</td>
</tr>
</tbody>
</table>
Table 3.7-3

Target PSD Compatible with Horizontal CSDRS

<table>
<thead>
<tr>
<th>Frequency ( f ) Range ( f ) (Hz or cps)</th>
<th>Piecewise Linear Target PSD ( S_H (f) ) (in^2/sec^4/cps)</th>
</tr>
</thead>
<tbody>
<tr>
<td>( 0.2 &lt; f \leq 2.5 ) Hz</td>
<td>( S_H (f) = 2\pi \times 58.4 \ (f / 2.5)^{0.2} )</td>
</tr>
<tr>
<td>( 2.5 &lt; f \leq 9.0 ) Hz</td>
<td>( S_H (f) = 2\pi \times 58.4 \ (2.5 / f)^{1.8} )</td>
</tr>
<tr>
<td>( 9.0 &lt; f \leq 16 ) Hz</td>
<td>( S_H (f) = 2\pi \times 5.82 \ (9.0 / f)^{2.2} )</td>
</tr>
<tr>
<td>( 16 &lt; f \leq 25 ) Hz</td>
<td>( S_H (f) = 2\pi \times 1.64 \ (16 / f)^{3.8} )</td>
</tr>
<tr>
<td>( 25 &lt; f \leq 35 ) Hz</td>
<td>( S_H (f) = 2\pi \times 0.30 \ (25 / f)^{4.8} )</td>
</tr>
<tr>
<td>( 35 &lt; f \leq 50 ) Hz</td>
<td>( S_H (f) = 2\pi \times 0.06 \ (35 / f)^{10} )</td>
</tr>
</tbody>
</table>
Table 3.7-4
Numerical Information of HRHF RS

a) Strong motion duration for all three directions (sec)

<table>
<thead>
<tr>
<th></th>
<th>H1H</th>
<th>H2H</th>
<th>VTH</th>
</tr>
</thead>
<tbody>
<tr>
<td>6.090</td>
<td>6.400</td>
<td>6.505</td>
<td></td>
</tr>
</tbody>
</table>

b) Peak values (A, V and D)

<table>
<thead>
<tr>
<th></th>
<th>A (g)</th>
<th>V (in/sec)</th>
<th>D (in)</th>
</tr>
</thead>
<tbody>
<tr>
<td>H1H</td>
<td>0.463</td>
<td>8.86</td>
<td>3.63</td>
</tr>
<tr>
<td>H2H</td>
<td>0.463</td>
<td>10.50</td>
<td>3.51</td>
</tr>
<tr>
<td>VTH</td>
<td>0.463</td>
<td>9.94</td>
<td>4.16</td>
</tr>
</tbody>
</table>

c) Number of points of the response spectra below CSDRS

<table>
<thead>
<tr>
<th></th>
<th>H1H</th>
<th>H2H</th>
<th>VTH</th>
</tr>
</thead>
<tbody>
<tr>
<td>2%</td>
<td>0</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>3%</td>
<td>2</td>
<td>4</td>
<td>1</td>
</tr>
<tr>
<td>4%</td>
<td>1</td>
<td>5</td>
<td>2</td>
</tr>
<tr>
<td>5%</td>
<td>0</td>
<td>4</td>
<td>3</td>
</tr>
<tr>
<td>7%</td>
<td>0</td>
<td>0</td>
<td>2</td>
</tr>
<tr>
<td>10%</td>
<td>4</td>
<td>3</td>
<td>2</td>
</tr>
</tbody>
</table>

d) The lowest percentage below CSDRS (%)

<table>
<thead>
<tr>
<th></th>
<th>H1H</th>
<th>H2H</th>
<th>VTH</th>
</tr>
</thead>
<tbody>
<tr>
<td>2%</td>
<td>1.02</td>
<td>-0.77</td>
<td>-2.94</td>
</tr>
<tr>
<td>3%</td>
<td>-1.85</td>
<td>-4.02</td>
<td>-1.49</td>
</tr>
<tr>
<td>4%</td>
<td>-0.67</td>
<td>-1.84</td>
<td>-1.61</td>
</tr>
<tr>
<td>5%</td>
<td>1.56</td>
<td>-3.02</td>
<td>-2.01</td>
</tr>
<tr>
<td>7%</td>
<td>0.76</td>
<td>0.33</td>
<td>-0.89</td>
</tr>
<tr>
<td>10%</td>
<td>-1.94</td>
<td>-1.09</td>
<td>-4.40</td>
</tr>
</tbody>
</table>
### Target PSD Compatible with Horizontal HRHF Seismic Input Motions

<table>
<thead>
<tr>
<th>Frequency Range ( f ) (Hz or cps)</th>
<th>Piecewise Linear Target PSD ( S_H(f) ) (in²/sec⁴/cps)</th>
</tr>
</thead>
<tbody>
<tr>
<td>( 0.3 &lt; f \leq 1.5 ) Hz</td>
<td>( S_0(f) = 2\pi \times 6.85 (0.3 / f)^{0.4} )</td>
</tr>
<tr>
<td>( 1.5 &lt; f \leq 4.0 ) Hz</td>
<td>( S_0(f) = 2\pi \times 13.04 (1.5 / f)^{0.2} )</td>
</tr>
<tr>
<td>( 4.0 &lt; f \leq 19 ) Hz</td>
<td>( S_0(f) = 2\pi \times 15.86 (4.0 / f)^{0.25} )</td>
</tr>
<tr>
<td>( 19 &lt; f \leq 40 ) Hz</td>
<td>( S_0(f) = 2\pi \times 10.75 (19.0 / f)^{1.1} )</td>
</tr>
<tr>
<td>( 40 &lt; f \leq 55 ) Hz</td>
<td>( S_0(f) = 2\pi \times 4.75 (40.0 / f)^{2.3} )</td>
</tr>
<tr>
<td>( 55 &lt; f \leq 70 ) Hz</td>
<td>( S_0(f) = 2\pi \times 2.28 (55.0 / f)^{4.5} )</td>
</tr>
<tr>
<td>( 70 &lt; f \leq 100 ) Hz</td>
<td>( S_0(f) = 2\pi \times 0.76 (70.0 / f)^{7.1} )</td>
</tr>
</tbody>
</table>
Table 3.7-6

Target PSD Compatible with Vertical HRHF Seismic Input Motions

<table>
<thead>
<tr>
<th>Frequency (f) Range $f$ (Hz or cps)</th>
<th>Piecewise Linear Target PSD $S_0(f)$ (in²/sec⁴/cps)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$0.3 &lt; f \leq 1.5$ Hz</td>
<td>$S_0(f) = 2\pi \times 3.44 \left(\frac{0.3}{f}\right)^{0.5}$</td>
</tr>
<tr>
<td>$1.5 &lt; f \leq 4.0$ Hz</td>
<td>$S_0(f) = 2\pi \times 7.69 \left(\frac{1.5}{f}\right)^{0.1}$</td>
</tr>
<tr>
<td>$4.0 &lt; f \leq 19$ Hz</td>
<td>$S_0(f) = 2\pi \times 8.49 \left(\frac{4.0}{f}\right)^{0.15}$</td>
</tr>
<tr>
<td>$19 &lt; f \leq 40$ Hz</td>
<td>$S_0(f) = 2\pi \times 6.72 \left(\frac{19.0}{f}\right)^{0.3}$</td>
</tr>
<tr>
<td>$40 &lt; f \leq 55$ Hz</td>
<td>$S_0(f) = 2\pi \times 5.38 \left(\frac{40.0}{f}\right)^{1.5}$</td>
</tr>
<tr>
<td>$55 &lt; f \leq 70$ Hz</td>
<td>$S_0(f) = 2\pi \times 3.34 \left(\frac{55.0}{f}\right)^{3.9}$</td>
</tr>
<tr>
<td>$70 &lt; f \leq 100$ Hz</td>
<td>$S_0(f) = 2\pi \times 1.31 \left(\frac{70.0}{f}\right)^{6.2}$</td>
</tr>
</tbody>
</table>
## Table 3.7-7 (1 of 2)

### Damping Values

<table>
<thead>
<tr>
<th>Category</th>
<th>Description</th>
<th>SSE</th>
<th>OBE</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Structural Material</strong></td>
<td>Welded steel or bolted steel with friction connections</td>
<td>4 %</td>
<td>3 %</td>
</tr>
<tr>
<td></td>
<td>Bolted steel with bearing connections</td>
<td>7 %</td>
<td>5 %</td>
</tr>
<tr>
<td></td>
<td>Prestressed concrete</td>
<td>5 %</td>
<td>3 %</td>
</tr>
<tr>
<td></td>
<td>Reinforced concrete</td>
<td>7 %</td>
<td>4 %</td>
</tr>
<tr>
<td><strong>Piping Systems</strong></td>
<td>Piping systems(1)</td>
<td>4 %</td>
<td>3 %</td>
</tr>
<tr>
<td><strong>Electrical Distribution Systems</strong></td>
<td>Full cable trays and related supports(2)</td>
<td>10 %</td>
<td>7 %</td>
</tr>
<tr>
<td></td>
<td>Empty cable trays and related supports</td>
<td>7 %</td>
<td>5 %</td>
</tr>
<tr>
<td></td>
<td>Full conduits and related supports</td>
<td>7 %</td>
<td>5 %</td>
</tr>
<tr>
<td></td>
<td>Empty conduits and related supports</td>
<td>5 %</td>
<td>3 %</td>
</tr>
<tr>
<td><strong>HVAC Duct Systems</strong></td>
<td>Pocket lock</td>
<td>10 %</td>
<td>7 %</td>
</tr>
<tr>
<td></td>
<td>Companion angle</td>
<td>7 %</td>
<td>5 %</td>
</tr>
<tr>
<td></td>
<td>Welded</td>
<td>4 %</td>
<td>3 %</td>
</tr>
<tr>
<td><strong>Mechanical and Electrical Components</strong></td>
<td>Motor, fan, compressor housings, pressure vessels, heat exchangers, pumps, and valve bodies</td>
<td>3 %</td>
<td>2 %</td>
</tr>
<tr>
<td></td>
<td>Electrical cabinets, panels, and MCCs</td>
<td>3 %</td>
<td>2 %</td>
</tr>
<tr>
<td></td>
<td>Welded instrument racks and tanks (impulsive mode) (3)</td>
<td>3 %</td>
<td>2 %</td>
</tr>
</tbody>
</table>
Table 3.7-7 (2 of 2)

(1) As an alternative for response spectrum analysis using an envelope of the SSE or OBE response spectra at all support points (uniform support motion), frequency-dependent damping values shown in the graph below may be used, subject to the following restrictions:

- Frequency-dependent damping should be used completely and consistently, if at all. Damping values for equipment other than piping are to be consistent with the values in the above table and NRC RG 1.61.
- Use of the specified damping values is limited only to response spectral analyses. Acceptance of the use of the specified damping values with other types of dynamic analyses (e.g., time-history analyses or independent support motion method) requires further justification.
- When used for reconciliation or support optimization of existing designs, the effects of increased motion on existing clearances and online mounted equipment should be checked.
- Frequency-dependent damping is not appropriate for analyzing the dynamic response of piping systems using supports designed to dissipated energy by yielding.
- Frequency-dependent damping is not applicable to piping in which stress corrosion cracking has occurred, unless a case-specific evaluation is provided, reviewed, and found acceptable by the NRC staff.

![Graph showing frequency-dependent damping values](image)

(2) The use of higher damping values for cable trays with flexible support systems (e.g., rod-hung trapeze systems, strut-hung trapeze systems, and strut-type cantilever and braced cantilever support systems) is permissible, subject to obtaining Regulatory Authority’s review for acceptance on a case-by-case basis.

(3) Use 0.5% damping for sloshing mode for tanks.
Table 3.7-8

Foundation Embedment Depth, Foundation Size, and Total Height of Seismic Category I Structures

<table>
<thead>
<tr>
<th>Structures</th>
<th>Foundation Embedment Depth, m (ft)</th>
<th>Foundation Size, m (ft)</th>
<th>Maximum Height, m (ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Nuclear Island</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>– Reactor Containment Building</td>
<td>See note 1.</td>
<td>Radius 25.6</td>
<td>87.9</td>
</tr>
<tr>
<td></td>
<td>16.4 (53’-8”)</td>
<td>(84’-0”)</td>
<td>(288’-6”)</td>
</tr>
<tr>
<td>– Auxiliary Building</td>
<td></td>
<td>107.3 × 88.1</td>
<td>56.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(352’-0” × 289’-0”)</td>
<td>(185’-0”)</td>
</tr>
<tr>
<td><strong>Emergency Diesel Generator (EDG) Building</strong></td>
<td>2.0 (6’-8”)</td>
<td>39.9 × 18.3</td>
<td>17.8</td>
</tr>
<tr>
<td><strong>Diesel Fuel Oil Tank (DFOT) Room</strong></td>
<td>12.1 (39’-8”)</td>
<td>20.3 × 18.3</td>
<td>18.7</td>
</tr>
</tbody>
</table>

(1) The auxiliary building wraps around the reactor containment building with a minimum of 6 inches seismic gap.
Table 3.7-9 (1 of 2)

Summary of Models and Analysis Methods

<table>
<thead>
<tr>
<th>Model</th>
<th>Analysis Method/Damping</th>
<th>Program</th>
<th>Type of Dynamic Response/Purpose</th>
<th>DCD/TeR Subsections</th>
</tr>
</thead>
</table>
| Reactor containment building fine-mesh model for seismic analysis (Uncracked stiffness model) | • Modal analysis  
• Direct integration time-history analysis  
• OBE damping | ANSYS | To verify the mesh sizes of the reactor containment building coarse-mesh model for seismic analysis | Technical Report APR1400-E-S-NR-14002-P, Section 3 |
| Auxiliary building fine-mesh model for seismic analysis (Uncracked stiffness model) | • Modal analysis  
• Direct integration time-history analysis  
• OBE damping | ANSYS | To verify the mesh sizes of the auxiliary building coarse-mesh model for seismic analysis | Technical Report APR1400-E-S-NR-14002-P, Section 4 |
| Reactor containment building coarse-mesh model for seismic analysis (Uncracked stiffness model) | • Modal analysis  
• Direct integration time-history analysis  
• OBE damping | ANSYS | To create and verify the SASSI reactor containment building model for seismic analysis | Technical Report APR1400-E-S-NR-14002-P, Section 3 |
| Auxiliary building coarse-mesh model for seismic analysis (Uncracked stiffness model) | • Modal analysis  
• Direct integration time-history analysis  
• OBE damping | ANSYS | To create and verify the SASSI auxiliary building model for seismic analysis | Technical Report APR1400-E-S-NR-14002-P, Section 4 |
| SASSI reactor containment building model for seismic analysis (Uncracked stiffness model) | • Complex frequency response analysis  
• OBE damping | ACS SASSI | To create the SASSI combined nuclear island model for seismic analysis | Technical Report APR1400-E-S-NR-14002-P, Section 5 |
| SASSI auxiliary building model for seismic analysis (Uncracked stiffness model) | • Complex frequency response analysis  
• OBE damping | ACS SASSI | To create the SASSI combined nuclear island model for seismic analysis | Technical Report APR1400-E-S-NR-14002-P, Section 5 |
### Table 3.7-9 (2 of 2)

<table>
<thead>
<tr>
<th>Model</th>
<th>Analysis Method/Damping</th>
<th>Program</th>
<th>Type of Dynamic Response/Purpose</th>
<th>DCD/TeR Subsections</th>
</tr>
</thead>
</table>
| SASSI combined nuclear island model for seismic analysis (Cracked and uncracked stiffness models) | • Complex frequency response analysis  
• OBE damping for uncracked stiffness model  
• SSE damping for cracked stiffness model | ACS SASSI | • To perform seismic analyses for eight generic soil profiles and one fixed-base case  
• To develop time histories for generating plant design in-structure response spectra  
• To obtain maximum absolute nodal acceleration  
• To obtain maximum displacements relative to the basemat and the free-field  
• To obtain maximum story shear forces and moments | Technical Report APR1400-E-S-NR-14002-P, Section 6 |
| EDG building (include DFOT room) model for seismic analysis (Uncracked stiffness model) | • Modal analysis  
• OBE damping | GTSTRUDL | • To create and verify the SASSI EDG building (include DFOT room) model for seismic analysis | Subsection 3.7.2.3.3.2 |
| SASSI EDG building (include DFOT room) model for seismic analysis (Cracked and uncracked stiffness models) | • Complex frequency response analysis  
• OBE damping for uncracked stiffness model  
• SSE damping for cracked stiffness model | ACS SASSI | • To perform seismic analyses for eight generic soil profiles and one fixed-base case  
• To develop time histories for generating plant design in-structure response spectra  
• To obtain maximum absolute nodal acceleration  
• To obtain maximum displacements relative to the basemat and the free-field  
• To obtain maximum story shear forces and moments | Subsection 3.7.2.3.3.2 |
<table>
<thead>
<tr>
<th>Modes</th>
<th>Natural Frequency (Hz)</th>
<th>Mass Ratio</th>
<th>Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td>11</td>
<td>3.49</td>
<td>18.2 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>12</td>
<td>3.58</td>
<td>15.7 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>30</td>
<td>10.60</td>
<td>22.5 %</td>
<td>Z (Vertical)</td>
</tr>
<tr>
<td>98</td>
<td>19.55</td>
<td>1.1 %</td>
<td>Drumming</td>
</tr>
</tbody>
</table>
### Summary of Modal Properties of Reactor Containment Building

#### Internal Structure

<table>
<thead>
<tr>
<th>Modes</th>
<th>Natural Frequency (Hz)</th>
<th>Mass Ratio</th>
<th>Direction</th>
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</thead>
<tbody>
<tr>
<td>18</td>
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<td>4.1 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>28</td>
<td>9.72</td>
<td>6.1 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>25</td>
<td>8.61</td>
<td>1.5 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>29</td>
<td>9.93</td>
<td>5.1 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>132</td>
<td>23.41</td>
<td>8.0 %</td>
<td>Z (Vertical)</td>
</tr>
</tbody>
</table>
### Summary of Modal Properties of Auxiliary Building

<table>
<thead>
<tr>
<th>Modes</th>
<th>Natural Frequency (Hz)</th>
<th>Mass Ratio</th>
<th>Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td>76</td>
<td>5.35</td>
<td>37.5 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>75</td>
<td>5.16</td>
<td>18.5 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>81</td>
<td>5.79</td>
<td>13.9 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>75</td>
<td>5.16</td>
<td>12.3 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>76</td>
<td>5.35</td>
<td>10.7 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>171</td>
<td>14.52</td>
<td>4.2 %</td>
<td>Z (VT)</td>
</tr>
<tr>
<td>140</td>
<td>12.65</td>
<td>3.8 %</td>
<td>Z (VT)</td>
</tr>
</tbody>
</table>
Table 3.7-13

Summary of Modal Properties of Emergency Diesel Generator Building

### EDG Building

<table>
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<tr>
<th>Modes</th>
<th>Natural Frequency (Hz)</th>
<th>Mass Ratio</th>
<th>Direction</th>
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</thead>
<tbody>
<tr>
<td>1</td>
<td>11.79</td>
<td>61.3 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>2</td>
<td>13.74</td>
<td>56.9 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>7</td>
<td>17.87</td>
<td>13.69 %</td>
<td>Z (Vertical)</td>
</tr>
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</table>

### DFOT Room

<table>
<thead>
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<th>Modes</th>
<th>Natural Frequency (Hz)</th>
<th>Mass Ratio</th>
<th>Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>13.35</td>
<td>31.87 %</td>
<td>X (E-W)</td>
</tr>
<tr>
<td>3</td>
<td>17.01</td>
<td>51.17 %</td>
<td>Y (N-S)</td>
</tr>
<tr>
<td>10</td>
<td>28.92</td>
<td>14.74 %</td>
<td>Z (Vertical)</td>
</tr>
</tbody>
</table>
Table 3.7-14

Maximum Response Accelerations of Reactor Containment Building

<table>
<thead>
<tr>
<th>Structure</th>
<th>Elevation (ft)</th>
<th>X</th>
<th>Y</th>
<th>Z</th>
</tr>
</thead>
<tbody>
<tr>
<td>Containment Structure</td>
<td>78</td>
<td>0.353</td>
<td>0.396</td>
<td>0.398</td>
</tr>
<tr>
<td></td>
<td>90</td>
<td>0.443</td>
<td>0.443</td>
<td>0.416</td>
</tr>
<tr>
<td></td>
<td>104</td>
<td>0.588</td>
<td>0.581</td>
<td>0.452</td>
</tr>
<tr>
<td></td>
<td>118</td>
<td>0.719</td>
<td>0.682</td>
<td>0.495</td>
</tr>
<tr>
<td></td>
<td>124</td>
<td>0.761</td>
<td>0.705</td>
<td>0.525</td>
</tr>
<tr>
<td></td>
<td>132</td>
<td>0.746</td>
<td>0.721</td>
<td>0.563</td>
</tr>
<tr>
<td></td>
<td>160</td>
<td>0.807</td>
<td>0.805</td>
<td>0.692</td>
</tr>
<tr>
<td></td>
<td>180</td>
<td>0.943</td>
<td>0.879</td>
<td>0.776</td>
</tr>
<tr>
<td></td>
<td>196</td>
<td>1.026</td>
<td>0.983</td>
<td>0.844</td>
</tr>
<tr>
<td></td>
<td>216</td>
<td>1.061</td>
<td>1.160</td>
<td>0.926</td>
</tr>
<tr>
<td></td>
<td>241</td>
<td>1.228</td>
<td>1.439</td>
<td>1.023</td>
</tr>
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<td>255</td>
<td>1.316</td>
<td>1.484</td>
<td>1.071</td>
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<td>275</td>
<td>1.406</td>
<td>1.522</td>
<td>1.127</td>
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<tr>
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<td>302</td>
<td>1.545</td>
<td>1.738</td>
<td>1.272</td>
</tr>
<tr>
<td></td>
<td>328</td>
<td>1.686</td>
<td>1.893</td>
<td>1.457</td>
</tr>
<tr>
<td></td>
<td>332</td>
<td>1.700</td>
<td>1.901</td>
<td>1.529</td>
</tr>
</tbody>
</table>
**Table 3.7-15**

**Maximum Relative Displacements of Reactor Containment Building**

**Containment Structure**

<table>
<thead>
<tr>
<th>Elevation (ft)</th>
<th>Maximum Displacements Relative to Basemat (in)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>X-direction</td>
</tr>
<tr>
<td>78</td>
<td>0.0906</td>
</tr>
<tr>
<td>90</td>
<td>0.1806</td>
</tr>
<tr>
<td>104</td>
<td>0.3372</td>
</tr>
<tr>
<td>118</td>
<td>0.4908</td>
</tr>
<tr>
<td>124</td>
<td>0.5520</td>
</tr>
<tr>
<td>132</td>
<td>0.6339</td>
</tr>
<tr>
<td>160</td>
<td>0.9273</td>
</tr>
<tr>
<td>180</td>
<td>1.1415</td>
</tr>
<tr>
<td>196</td>
<td>1.3047</td>
</tr>
<tr>
<td>216</td>
<td>1.5239</td>
</tr>
<tr>
<td>241</td>
<td>1.7880</td>
</tr>
<tr>
<td>255</td>
<td>1.9243</td>
</tr>
<tr>
<td>275</td>
<td>2.1183</td>
</tr>
<tr>
<td>302</td>
<td>2.3529</td>
</tr>
<tr>
<td>332</td>
<td>2.6010</td>
</tr>
</tbody>
</table>
Table 3.7-16

Maximum Member Forces of Reactor Containment Building
Containment Structure

<table>
<thead>
<tr>
<th>Elevation (ft)</th>
<th>Fx</th>
<th>Fy</th>
<th>Fz</th>
<th>Mx</th>
<th>My</th>
<th>Mz</th>
</tr>
</thead>
<tbody>
<tr>
<td>307.5</td>
<td>1.195E+04</td>
<td>1.224E+04</td>
<td>1.169E+04</td>
<td>1.762E+05</td>
<td>1.779E+05</td>
<td>1.130E+04</td>
</tr>
<tr>
<td>281</td>
<td>2.305E+04</td>
<td>2.365E+04</td>
<td>2.187E+04</td>
<td>7.281E+05</td>
<td>7.241E+05</td>
<td>3.730E+04</td>
</tr>
<tr>
<td>254.5</td>
<td>3.047E+04</td>
<td>3.147E+04</td>
<td>2.887E+04</td>
<td>1.562E+06</td>
<td>1.541E+06</td>
<td>6.021E+04</td>
</tr>
<tr>
<td>220</td>
<td>4.543E+04</td>
<td>4.722E+04</td>
<td>4.172E+04</td>
<td>3.222E+06</td>
<td>3.116E+06</td>
<td>1.034E+05</td>
</tr>
<tr>
<td>200</td>
<td>5.117E+04</td>
<td>5.327E+04</td>
<td>4.842E+04</td>
<td>4.326E+06</td>
<td>4.184E+06</td>
<td>1.247E+05</td>
</tr>
<tr>
<td>178</td>
<td>5.587E+04</td>
<td>5.823E+04</td>
<td>5.461E+04</td>
<td>5.634E+06</td>
<td>5.456E+06</td>
<td>1.432E+05</td>
</tr>
<tr>
<td>156</td>
<td>5.959E+04</td>
<td>6.205E+04</td>
<td>6.000E+04</td>
<td>7.023E+06</td>
<td>6.798E+06</td>
<td>1.640E+05</td>
</tr>
<tr>
<td>136</td>
<td>6.163E+04</td>
<td>6.488E+04</td>
<td>6.357E+04</td>
<td>8.315E+06</td>
<td>8.054E+06</td>
<td>1.779E+05</td>
</tr>
<tr>
<td>125</td>
<td>6.255E+04</td>
<td>6.609E+04</td>
<td>6.489E+04</td>
<td>9.037E+06</td>
<td>8.747E+06</td>
<td>1.837E+05</td>
</tr>
<tr>
<td>100</td>
<td>6.660E+04</td>
<td>7.086E+04</td>
<td>6.908E+04</td>
<td>1.071E+07</td>
<td>1.036E+07</td>
<td>1.989E+05</td>
</tr>
<tr>
<td>78</td>
<td>6.795E+04</td>
<td>7.249E+04</td>
<td>7.098E+04</td>
<td>1.223E+07</td>
<td>1.177E+07</td>
<td>2.037E+05</td>
</tr>
</tbody>
</table>
### Table 3.7-17

**Maximum Response Accelerations of Reactor Containment Building Internal Structure**

<table>
<thead>
<tr>
<th>Structure</th>
<th>Elevation (ft)</th>
<th>Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>X</td>
</tr>
<tr>
<td><strong>Primary Shield Wall</strong></td>
<td>66</td>
<td>0.333</td>
</tr>
<tr>
<td></td>
<td>78</td>
<td>0.360</td>
</tr>
<tr>
<td></td>
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<td>0.413</td>
</tr>
<tr>
<td></td>
<td>107</td>
<td>0.433</td>
</tr>
<tr>
<td></td>
<td>114</td>
<td>0.504</td>
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<td></td>
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<td>0.488</td>
</tr>
<tr>
<td></td>
<td>137</td>
<td>0.539</td>
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<tr>
<td></td>
<td>156</td>
<td>1.011</td>
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<tr>
<td></td>
<td>191a</td>
<td>1.218</td>
</tr>
<tr>
<td></td>
<td>191b</td>
<td>1.647</td>
</tr>
<tr>
<td><strong>Secondary Shield Wall</strong></td>
<td>78</td>
<td>0.350</td>
</tr>
<tr>
<td></td>
<td>100a</td>
<td>0.459</td>
</tr>
<tr>
<td></td>
<td>100b</td>
<td>0.520</td>
</tr>
<tr>
<td></td>
<td>107</td>
<td>0.543</td>
</tr>
<tr>
<td></td>
<td>114</td>
<td>0.613</td>
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<tr>
<td></td>
<td>130</td>
<td>0.631</td>
</tr>
<tr>
<td></td>
<td>137</td>
<td>0.690</td>
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<tr>
<td></td>
<td>156</td>
<td>1.182</td>
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<tr>
<td></td>
<td>191</td>
<td>2.524</td>
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<tr>
<td><strong>Slabs</strong></td>
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<td>-</td>
</tr>
<tr>
<td></td>
<td>156</td>
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</tr>
</tbody>
</table>

191a: Primary shield wall top  
191b: Pressurizer room corners  
100a: Secondary shield wall interface with concrete pedestal top  
100b: In-containment refueling water storage tank walls
### Maximum Relative Displacements of Reactor Containment Building Internal Structure

<table>
<thead>
<tr>
<th>Structure</th>
<th>Elevation (ft)</th>
<th>X-direction</th>
<th>Y-direction</th>
<th>Z-direction</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary Shield Wall</td>
<td>66</td>
<td>0.0396</td>
<td>0.0435</td>
<td>0.1305</td>
</tr>
<tr>
<td></td>
<td>78</td>
<td>0.0853</td>
<td>0.0906</td>
<td>0.1310</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>0.1741</td>
<td>0.1825</td>
<td>0.1613</td>
</tr>
<tr>
<td></td>
<td>107</td>
<td>0.2015</td>
<td>0.2103</td>
<td>0.1670</td>
</tr>
<tr>
<td></td>
<td>114</td>
<td>0.2337</td>
<td>0.2449</td>
<td>0.1685</td>
</tr>
<tr>
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<td>130</td>
<td>0.2952</td>
<td>0.3119</td>
<td>0.1701</td>
</tr>
<tr>
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<td>137</td>
<td>0.3268</td>
<td>0.3499</td>
<td>0.1616</td>
</tr>
<tr>
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## Maximum Member Forces of Reactor Containment Building
### Internal Structure

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<th>Fz</th>
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Table 3.7-20

Maximum Response Accelerations of Auxiliary Building

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<td>Y</td>
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<td>0.314</td>
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<td>1.499</td>
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<tr>
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195a: Main control room roof
195b: Fuel handling area
195c: Control area 2 emergency exhaust ACU and normal exhaust ACU
## Table 3.7-21

### Maximum Relative Displacements of Auxiliary Building

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<th>Floor Label</th>
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<th>Z-direction (Wall)</th>
<th>Z-direction (Slab)</th>
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<td>1-M</td>
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<td>0.990</td>
<td>0.2790</td>
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### Table 3.7-22

Maximum Member Forces of Auxiliary Building

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<th>Fy</th>
<th>Fz</th>
<th>Mx</th>
<th>My</th>
<th>Mz</th>
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<td>213.5(^{(1)})</td>
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<td>7174</td>
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\(^{(1)}\) East part of Auxiliary Building  
\(^{(2)}\) West part of Auxiliary Building
Table 3.7-23

Maximum Response Accelerations of EDG Building

**EDG Building**

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<th>E-W (g)</th>
<th>N-S (g)</th>
<th>Vertical (g)</th>
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**DFOT Room**

<table>
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<th>Slab Elevation</th>
<th>E-W (g)</th>
<th>N-S (g)</th>
<th>Vertical (g)</th>
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### Table 3.7-24

**Maximum Relative Displacements of EDG Building**

#### EDG Building

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<th>Elevation</th>
<th>E-W Direction (in)</th>
<th>N-S Direction (in)</th>
<th>Vertical Direction (in)</th>
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#### DFOT Room

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<th>N-S Direction (in)</th>
<th>Vertical Direction (in)</th>
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Table 3.7-25

Maximum Member Forces of EDG Building

EDG Building

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<th>Fz</th>
<th>Mx</th>
<th>My</th>
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DFOT Room

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<th>Fy</th>
<th>Fz</th>
<th>Mx</th>
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<td>39058</td>
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Figure 3.7-1  Horizontal CSDRS
Figure 3.7-2  Vertical CSDRS
Figure 3.7-3  Acceleration, Velocity, and Displacement of H1 Design Ground Motion
Figure 3.7-4  Acceleration, Velocity, and Displacement of H2 Design Ground Motion
Figure 3.7-5  Acceleration, Velocity, and Displacement of VT Design Ground Motion
Figure 3.7-6  Comparison of Response Spectra of H1 Design Ground Motion and CSDRS
Figure 3.7-7  Comparison of Response Spectra of H2 Design Ground Motion and CSDRS
Figure 3.7-8  Comparison of Response Spectra of VT Design Ground Motion and CSDRS
(a) Comparison of PSD of H1 Design Ground Motion and Target PSD and Minimum Required PSD

(b) Comparison of PSD for H1 Design Ground Motion and PSD Consistent with SRP 3.7.1, Rev.4, Appendix B

Figure 3.7-9  Comparison of PSD for H1 Design Ground Motion
APR1400 DCD TIER 2

(a) Comparison of PSD of H2 Design Ground Motion and Target PSD and Minimum Required PSD

(b) Comparison of PSD for H2 Design Ground Motion and PSD Consistent with SRP 3.7.1, Rev.4, Appendix B

Figure 3.7-10 Comparison of PSD for H2 Design Ground Motion
(a) Comparison of PSD of VT Design Ground Motion and Target PSD and Minimum Required PSD

(b) Comparison of PSD for VT Design Ground Motion and PSD Consistent with SRP 3.7.1, Rev. 4, Appendix B

Figure 3.7-11  Comparison of PSD for VT Design Ground Motion
Figure 3.7-12  HRHF Horizontal Target Response Spectra
Figure 3.7-13  HRHF Vertical Target Response Spectra
Figure 3.7-14  Acceleration, Velocity, and Displacement of H1H for HRHF
Figure 3.7-15  Acceleration, Velocity, and Displacement of H2H for HRHF
Figure 3.7-16  Acceleration, Velocity, and Displacement of VTH for HRHF
Figure 3.7-17  Comparison of Response Spectra of H1H and HRHF Horizontal Target Design Spectra
Figure 3.7-18  Comparison of Response Spectra of H2H and HRHF Horizontal Target Design Spectra
Figure 3.7-19  Comparison of Response Spectra of VTH and HRHF Vertical Target Design Spectra
(a) Comparison of PSD of H1H and HRHF Target PSD and Minimum Required PSD

(b) Comparison of PSD for H1H and PSD Consistent with SRP 3.7.1, Rev.4, Appendix B

Figure 3.7-20  Comparison of PSD for H1H
(a) Comparison of PSD of H2H and HRHF Target PSD and Minimum Required PSD

(b) Comparison of PSD for H2H and PSD Consistent with SRP 3.7.1, Rev.4, Appendix B

Figure 3.7-21  Comparison of PSD for H2H
(a) Comparison of PSD of VTH and HRHF Target PSD and Minimum Required PSD

(b) Comparison of PSD for VTH and PSD Consistent with SRP 3.7.1, Rev.4, Appendix B

Figure 3.7-22  Comparison of PSD for VTH
Figure 3.7-23  Generic Soil Profiles Proposed for APR1400 Standard Design
Figure 3.7-24  Finite Element Model of Reactor Containment Building Containment Structure
Figure 3.7-25  Finite Element Model of Reactor Containment Building Internal Structure
Figure 3.7-26  Finite Element Model of Auxiliary Building
Figure 3.7-27  Finite Element Model of Emergency Diesel Generator Building
Figure 3.7-28  First Major X-mode Shape of Reactor Containment Building
Containment Structure
Figure 3.7-29  First Major Y-mode Shape of Reactor Containment Building
Containment Structure
Figure 3.7-30  First Major Z-mode Shape of Reactor Containment Building Containment Structure
Figure 3.7-31  First Major X-mode Shape of Reactor Containment Building Internal Structure
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APPENDIX 3.7A

SOIL-STRUCTURE INTERACTION ANALYSIS
METHODOLOGY AND RESULTS
# APPENDIX 3.7A – SOIL-STRUCTURE INTERACTION ANALYSIS

## METHODOLOGY AND RESULTS

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APPENDIX 3.7A – SOIL-STRUCTURE INTERACTION ANALYSIS

METHODOLOGY AND RESULTS

3.7A.1 Overview

This appendix presents the analysis results that are used to establish the seismic design loads for the nuclear island structures and the emergency diesel generator building in the APR1400 standard plant design. The soil-structure interaction (SSI) analysis and fixed-base analysis are performed using the seismic input parameters and methodology described in Section 3.7.

Three-dimensional SSI analyses are performed based on a safe shutdown earthquake (SSE) excitation of 0.3g horizontal and vertical peak ground accelerations. A set of the certified seismic design response spectra (CSDRS)-compatible time histories is used as the input excitation. The spectral characteristics of the input motions are described in Subsection 3.7.1. A set of eight soil profiles developed to represent generic site conditions are used as the soil medium in the SSI analysis. The SSI analysis results are provided in the form of in-structure response spectra corresponding to major elevations and internal resisting forces on each floor of seismic Category I structures.

A fixed-base analysis with no SSI effects is also performed using CSDRS-compatible time histories. The model for the fixed-base analysis is identical to the model for the SSI analysis except that the shear and compression wave velocities of the foundation medium are set to high values as 6,096 m/sec (20,000 ft/sec) for shear and 15,240 m/sec (50,000 ft/sec) for compression wave velocities and the backfill soil elements are removed so there is no contact of side soils with the structures over the embedment depth. The prescribed seismic input motion on the surface of the finished grade is the same as for the other soil cases. Because the foundation medium is stiff, there is virtually no variation in ground motion with depth. Therefore, the seismic input motion is considered the same as the prescribed motion at the bottom of the basemat level.

This appendix also describes the SSI analyses that are performed for seismic Category I structures. The SSI analyses use the soil profiles described in Subsection 3.7.1.3 and the control motions and structural models described in Subsections 3.7.1.1.2 and 3.7.2.3.3, respectively. The SSI analyses and fixed-base analysis use the ACS SASSI computer program (Reference 1). The final analysis results are obtained by enveloping the SSI analysis results and the fixed-base analysis results.
3.7A.2 Free-field Seismic Response Motions

Free-field response motions at the foundation base elevation are obtained from the site response analyses. The soil models using the generic soil/rock properties as shown in Tables 3.7A-1 through 3.7A-8 are then subjected to the control motion input defined as outcrop motion at the site grade elevation.

Figures 3.7A-9 through 3.7A-11 show the comparison of 5 percent damped response spectra between the seismic input motion at the free-field site grade elevation and the seismic response outcrop motions enveloped for all site soil column models at the seismic Category I structures foundation base elevation. The peak ground accelerations of the horizontal site response motions at the seismic Category I structures foundation base elevation are greater than 0.1g.

3.7A.3 SSI Analysis

Because the analyses are performed in the frequency domain, the transfer functions are generated up to a maximum cut-off frequency. Cut-off frequencies are the maximum frequencies that the soil media can transmit without loss of accuracy in the solution. In the present analyses, cut-off frequencies are computed based on the dimensions of the soil discretization. The maximum frequency that a soil layer can transmit corresponds to a wavelength equal to 5h, where h is the layer thickness. Cut-off frequencies vary according to the soil profiles used in the analyses. Table 3.7A-9 lists the cut-off frequencies for eight soil profiles.

3.7A.3.1 SSI Analysis Cases

A summary of the SSI analysis cases is presented in Table 3.7A-9. Eight SSI analyses are performed using all generic soil profiles described in Subsection 3.7.1.3.

All analyses are three-dimensional with input excitation provided in three directions. The generic soil sites differ from each other with respect to soil properties and depth of soil over bedrock. The embedment depth of the nuclear island is approximately 16.4 m (53 ft 8 in) in all cases.

The soil layers used in the SSI models and their associated generic soil/rock properties are shown in Tables 3.7A-1 to 3.7A-8 for all soil cases.
3.7A.3.2 Generation of Acceleration Time Histories

Acceleration time histories at key locations for the horizontal and vertical analyses due to the three control motions are computed. The locations selected are summarized in Subsection 3.7A.3.4. Time histories are computed for X, Y, and Z translations as follows:

a. The module MOTION is first used to generate transfer functions in the frequency domain at key locations. MOTION uses the uninterpolated transfer functions at the computed frequencies from ANALYS and performs the interpolation for intermediate frequencies. The transfer functions are computed up to the cut-off frequency. The interpolated transfer functions are reviewed to provide reasonable assurance that spurious spikes are not caused by the interpolation. At this step, the control motions are not yet applied.

b. When the transfer functions are finalized, MOTION is again executed for the same set of frequencies as in step (a) but this time multiplying the control motions with the transfer functions in the frequency domain and then obtaining acceleration time histories in the time domain through an inverse Fourier transform technique. In this step, MOTION convolutes the control motions and outputs results in the time domain. Three sets of MOTION runs are made for X, Y, and Z directions, using the appropriate transfer function file in each direction with output at identical locations. This step is repeated three times for control motions in X, Y, and Z directions. All control motions are for a duration of 20.48 seconds at a time step of 0.005 second.

3.7A.3.3 Generation of In-structure Response Spectra

The generation of in-structure response spectra (ISRS) first involves the computation of acceleration time histories at selected in-structure locations X(i), Y(i), and Z(i), i = x, y, z, as described in the previous step.

The time histories X(i), Y(i), and Z(i), i = x, y, z at each location are then used as input for the computation of ISRS for 2, 3, 4, 5, 7, and 10 percent damping. The response spectra are computed at numerous frequencies. These frequencies cover the frequency range of interest and exceed the guidelines of SRP 3.7.1. The total ISRS at each location is then computed using the square-root-of-the-sum-of-the-squares (SRSS) method. The ISRS are
then widened by ±15 percent in accordance with NRC Regulatory Guide (RG) 1.122 (Reference 2).

3.7A.3.4 Output Locations

Table 3.7A-10 shows the key locations where the ISRS are obtained. The key locations are locations that are expected to represent the minimum and maximum seismic responses in the structure and include the basemat and roof slab elevations and the support elevations of the major equipment.

3.7A.4 Analysis Results

Representative plots of the envelope SSE response spectra are presented in Figures 3.7A-12 to 3.7A-90. The corresponding locations are at the foundation basemat (El. 55 ft 0 in), auxiliary building (El. 100 ft 0 in, 156 ft 0 in), top of the auxiliary building (El. 213 ft 0 in, 216 ft 9 in), reactor containment building containment structure (El. 104 ft 0 in, 160 ft 0 in), top of the reactor containment building containment structure dome (El. 332 ft 0 in), reactor containment building internal structure (El. 78 ft 0 in, 156 ft 0 in), and top of reactor containment building internal structure dome (El. 191 ft 0 in), as described in Table 3.7A-10. Damping ratios are 2, 3, 4, 5, 7, and 10 percent of critical. Responses from all soil cases are superimposed on the same plot for each location and each direction (X, Y, Z). The fixed-base spectra are also superimposed on these plots for completeness of the envelope.

The range of site parameters used in the SSI analyses covers a broad range of site conditions. Soil amplification occurs at frequencies in the range of the dominant structural modal frequencies. Therefore, resonance effects between the soil and the structures are captured in the SSI analyses and are reflected in the results. As such, the combined SSI results provide reasonable assurance that adequate seismic loads for APR1400 seismic Category I structures have been generated for sites that are compatible with the generic sites used in these analyses.

3.7A.5 References

Table 3.7A-1 (1 of 4)

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Table 3.7A-1 (4 of 4)

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(1) Unit weight density of soil/rock
(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.
(3) Poisson’s Ratio
### Table 3.7A-2 (1 of 4)

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Table 3.7A-2 (4 of 4)

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(1) Unit weight density of soil/rock
(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.
(3) Poisson’s Ratio
### APR1400 DCD TIER 2

#### Table 3.7A-3 (1 of 4)

**Soil Layers and Properties (S3)**

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### APR1400 DCD TIER 2

#### Table 3.7A-3 (4 of 4)

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(1) Unit weight density of soil/rock  
(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.  
(3) Poisson’s Ratio
### Soil Layers and Properties (S4)

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### Table 3.7A-4 (2 of 4)

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(1) Unit weight density of soil/rock

(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.

(3) Poisson’s Ratio
Table 3.7A-5 (1 of 4)

Soil Layers and Properties (S6)

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### Table 3.7A-5 (3 of 4)

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Soil Type | Layer No. | Thick. (ft) | $\gamma^{(1)}$ (k/ft³) | Damp. | $V_s$ (ft/s) | $V_p^{(2)}$ (ft/s) | $\rho^{(3)}$
--- | --- | --- | --- | --- | --- | --- | ---
Rock (cont.) | 91 | 20 | 0.155 | 0.010 | 9,200 | 18,264 | 0.33
92 | 20 | 0.155 | 0.010 | 9,200 | 18,264 | 0.33
93 | 20 | 0.155 | 0.010 | 9,200 | 18,264 | 0.33
94 | 20 | 0.155 | 0.010 | 9,200 | 18,264 | 0.33
95 | 20 | 0.155 | 0.010 | 9,200 | 18,264 | 0.33
96 | – | 0.155 | 0.004 | 9,200 | 18,264 | 0.33

(1) Unit weight density of soil/rock
(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.
(3) Poisson’s Ratio
# Table 3.7A-6 (1 of 4)

## Soil Layers and Properties (S7)

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<th>Damp.</th>
<th>Vs (ft/s)</th>
<th>Vp(^{(2)} ) (ft/s)</th>
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<th>Damp.</th>
<th>Vs (ft/s)</th>
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1. Unit weight density of soil/rock
2. A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.
3. Poisson’s Ratio
Table 3.7A-7 (1 of 4)

Soil Layers and Properties (S8)

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### Table 3.7A-7 (2 of 4)

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(1) Unit weight density of soil/rock
(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.
(3) Poisson’s Ratio
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### Table 3.7A-8 (4 of 4)

<table>
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<tr>
<th>Soil Type</th>
<th>Layer No.</th>
<th>Thick. (ft)</th>
<th>$\gamma^{(1)}$ (k/ft$^3$)</th>
<th>Damp.</th>
<th>$V_S$ (ft/s)</th>
<th>$V_P^{(2)}$ (ft/s)</th>
<th>$\rho^{(3)}$</th>
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(1) Unit weight density of soil/rock

(2) A minimum compression wave velocity of 1,463 m/sec (4,800 ft/sec) (speed of sound in water) is used to consider groundwater table.

(3) Poisson’s Ratio
### Table 3.7A-9

SSI Analysis Cases and Cut-off Frequencies

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<thead>
<tr>
<th>Layer</th>
<th>S1</th>
<th>S2</th>
<th>S3</th>
<th>S4</th>
<th>S6</th>
<th>S7</th>
<th>S8</th>
<th>S9</th>
<th>S10</th>
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<tr>
<td>Above Basemat(1)</td>
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<td>40</td>
<td>34</td>
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<td>34</td>
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<td>381</td>
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<tr>
<td>Below Basemat(2)</td>
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<td>42</td>
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<td>77</td>
<td>173</td>
<td>150</td>
<td>263</td>
<td>193</td>
<td>800</td>
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</table>

(1) Top of soil profile, min. with upper nuclear island basemat

(2) General profile under nuclear island basemat
Table 3.7A-10

**Output Locations**

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<th>Structures</th>
<th>Elevation</th>
<th>Remarks</th>
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<td>78’-0”</td>
<td>Basemat elevation</td>
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<tr>
<td></td>
<td>100’-0”</td>
<td>Support elevation of major equipment</td>
</tr>
<tr>
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<td>156’-0”</td>
<td>Operation floor elevation</td>
</tr>
<tr>
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<td>191’-0”</td>
<td>Top of internal structure</td>
</tr>
<tr>
<td>Reactor Containment Building Containment Structure</td>
<td>104’-0”</td>
<td>Ground floor elevation</td>
</tr>
<tr>
<td></td>
<td>160’-0”</td>
<td>Operation floor elevation</td>
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<td></td>
<td>332’-0”</td>
<td>Top of dome</td>
</tr>
<tr>
<td>Auxiliary Building</td>
<td>55’-0”</td>
<td>Basemat elevation</td>
</tr>
<tr>
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<td>100’-0”</td>
<td>Ground floor elevation</td>
</tr>
<tr>
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<td>156’-0”</td>
<td>Main control room floor elevation</td>
</tr>
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<td>213’-0”</td>
<td>Roof of auxiliary building (1)</td>
</tr>
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<td>216’-9”</td>
<td>Roof of auxiliary building (2)</td>
</tr>
<tr>
<td>Emergency Diesel Generator Building</td>
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<td>Foundation basemat</td>
</tr>
<tr>
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<td>135’-0”</td>
<td>Roof slab elevation</td>
</tr>
<tr>
<td>Diesel Fuel Oil Storage Tank Room</td>
<td>63’-0”</td>
<td>Foundation basemat</td>
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<td>100’-0”</td>
<td>Roof slab elevation</td>
</tr>
</tbody>
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Figure 3.7A-1  Shear and Compression Wave Velocity Profiles for S1
Figure 3.7A-2  Shear and Compression Wave Velocity Profiles for S2
Figure 3.7A-3  Shear and Compression Wave Velocity Profiles for S3
Figure 3.7A-4 Shear and Compression Wave Velocity Profiles for S4
Figure 3.7A-5  Shear and Compression Wave Velocity Profiles for S6
Figure 3.7A-6  Shear and Compression Wave Velocity Profiles for S7
Figure 3.7A-7 Shear and Compression Wave Velocity Profiles for S8
Figure 3.7A-8  Shear and Compression Wave Velocity Profiles for S9
Figure 3.7A-9  Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Nuclear Island Structure, E-W Motion, 5% damping (1 of 3)
Figure 3.7A-9  Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Nuclear Island Structure, N-S Motion, 5% damping (2 of 3)
Figure 3.7A-9  Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Nuclear Island Structure, Vertical Motion, 5% damping (3 of 3)
Figure 3.7A-10  Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Emergency Diesel Generator Building, E-W Motion, 5% damping (1 of 3)
Figure 3.7A-10  Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Emergency Diesel Generator Building, N-S Motion, 5% damping (2 of 3)
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Figure 3.7A-11 Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Diesel Fuel Oil Tank, E-W Motion, 5% damping (1 of 3)
Figure 3.7A-11  Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Diesel Fuel Oil Tank, N-S Motion, 5% damping (2 of 3)
Figure 3.7A-11 Comparison of Design Time History Compatible with CSDRS with Site Response Outcrop Motion at Foundation Base Elevation of Diesel Fuel Oil Tank, Vertical Motion, 5% damping (3 of 3)
Figure 3.7A-12 Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 78'-0", E-W Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-13  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 78'-'0", N-S Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-14 Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 78'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-15 Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 100'-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-16  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 100'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-17  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 100'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-18  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 156'0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-19  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 156'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-20  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 156'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-21  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 191’-0” (General), E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-22  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 191'-0" (General), N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
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Figure 3.7A-24 Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 191'-0" (Pressurizer Corners), E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-25  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 191'-0" (Pressurizer Corners), N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-26  Enveloped ISRS for SSE, Reactor Containment Building PSW at El. 191'-0" (Pressurizer Corners), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-27 Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 78'-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-28  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 78'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
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Figure 3.7A-31  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 100'-0" (General), N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-32  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 100'-0" (General), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-33 Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 100'-0" (IRWST Walls), E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-34  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 100'-0" (IRWST Walls), N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-35  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 100'-0" (IRWST Walls), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-36  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 156'-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-37 Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 156'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-38  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 156'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-39  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 191’-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-40  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 191'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-41  Enveloped ISRS for SSE, Reactor Containment Building SSW at El. 191'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-42  Enveloped ISRS for SSE, Reactor Containment Building  
Containment Structure at El. 104'-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-43 Enveloped ISRS for SSE, Reactor Containment Building
Containment Structure at El. 104'0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-44 Enveloped ISRS for SSE, Reactor Containment Building
Containment Structure at El. 104'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-45 Enveloped ISRS for SSE, Reactor Containment Building
Containment Structure at El. 160’-0”, E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-46 Enveloped ISRS for SSE, Reactor Containment Building
Containment Structure at El. 160'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-47  Enveloped ISRS for SSE, Reactor Containment Building
Containment Structure at El. 160'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-48  Enveloped ISRS for SSE, Reactor Containment Building
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Figure 3.7A-49 Enveloped ISRS for SSE, Reactor Containment Building Containment Structure at El. 332'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-50  Enveloped ISRS for SSE, Reactor Containment Building
Containment Structure at El. 332'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-51 Enveloped ISRS for SSE, Auxiliary Building at El. 55'-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-52  Enveloped ISRS for SSE, Auxiliary Building at El. 55'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-53  Enveloped ISRS for SSE, Auxiliary Building at El. 55'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-54  Enveloped ISRS for SSE, Auxiliary Building at El. 55’-0” (Slab), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-55  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (General), E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-56  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (General), N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-57  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (General), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-58 Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (General Slab), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-59  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (Fuel Handling Area), E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-60  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (Fuel Handling Area), N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-61  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (Fuel Handling Area), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-62  Enveloped ISRS for SSE, Auxiliary Building at El. 100'-0" (Fuel Handling Area Slab), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-63 Enveloped ISRS for SSE, Auxiliary Building at El. 156'-0", E-W, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-64  Enveloped ISRS for SSE, Auxiliary Building at El. 156'-0", N-S, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-65  Enveloped ISRS for SSE, Auxiliary Building at El. 156'-0", Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
Figure 3.7A-66  Enveloped ISRS for SSE, Auxiliary Building at El. 156'-0" (Slab), Vertical, Damping Ratio 2%, 3%, 4%, 5%, 7%, 10%
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APPENDIX 3.7B

EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT
# APPENDIX 3.7B – EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

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APPENDIX 3.7B – EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

3.7B.1 Overview

The seismic analysis and design of the APR1400 standard plant are based on the certified seismic design response spectra (CSDRS) described in Subsection 3.7.1.1.1. These spectra are based on NRC Regulatory Guide (RG) 1.60 (Reference 1) with an increase in response spectrum amplitude in the 9 to 50 Hz frequency range. However, many of the envelope response spectra of the Central and Eastern United States (CEUS) rock sites show higher response spectrum amplitudes at higher frequencies than the CSDRS. Response spectra with these characteristics are called herein as the hard rock high frequency (HRHF) ground motion response spectra (GMRS).

The HRHF response spectra considered for the APR1400 standard design are shown in Figures 3.7B-1 and 3.7B-2 for the horizontal and vertical directions, respectively, where they are also compared to the APR1400 CSDRS for both the horizontal and vertical directions for 5 percent damping. The HRHF response spectra exceed the CSDRS for frequencies above approximately 10 Hz. However, as presented in the EPRI Draft White Paper, “Considerations for NPP Equipment and Structures Subjected to Response Levels Caused by High Frequency Ground Motions” (Reference 2), the high frequency seismic input is regarded generally as non-damaging.

The structures and equipment qualified by seismic analyses for the seismic response to APR1400 CSDRS are evaluated for the potential to sustain damage from high frequency seismic input motion. The building structures, reactor pressure vessel and internals, primary component supports, primary loop nozzles, piping, and equipment were included in this evaluation to demonstrate that the seismic responses of the selected structures, components, supports, and piping subjected to the high frequency seismic input motion are non-damaging. In the evaluation, the seismic responses of the structures due to high frequency seismic input motion have considered the effects of spatial incoherence of seismic input motion.

This appendix summarizes the methodology and results of the evaluation of the effects of HRHF response spectra the selected on structures, systems, and components (SSCs) of the APR1400 standard plant. The detailed methodology and results of the evaluation are documented in Technical Report, APR1400-E-S-NR-14004-P (Reference 3).
3.7B.2  High Frequency Seismic Input

Comparisons of the horizontal and vertical HRHF response spectra considered for the evaluation of the APR1400 design with the APR1400 CSDRS are presented in Figures 3.7B-1 and 3.7B-2. The APR1400 HRHF response spectra used for the evaluation were developed for the CEUS hard-rock sites. These spectra are higher than the CSDRS in the high frequency range, from approximately 10 to 100 Hz.

The EPRI report, “Evaluation of Seismic Hazard at Central and Eastern US Nuclear Power Plant Sites” (Reference 4), was used to develop the APR1400 HRHF response spectra. In the EPRI report, the maximum and fractile GMRS for the 60 CEUS nuclear power plant sites, assuming they are all hard-rock sites, were developed. The 0.8-fractile, 5 percent damped, horizontal composite envelope GMRS for CEUS hard-rock sites to achieve the appropriate goal of non-exceedance probability were selected as the HRHF horizontal target response spectrum for the APR1400 standard plant evaluation, as shown in Figure 3.7B-3.

The 5 percent damped HRHF vertical target response spectrum was generated by multiplying the vertical/horizontal (V/H) ratios for CEUS rock sites given in Table 4-5 of NUREG/CR-6728 (Reference 5) by the 5 percent damped HRHF horizontal target response spectrum. The resulting 5 percent damped HRHF vertical target response spectrum that was generated is plotted in Figure 3.7B-4.

The digitized values of the HRHF horizontal and vertical target response spectra are given in Tables 3.7B-1 and 3.7B-2, respectively.

The HRHF horizontal target response spectra for damping ratios other than 5 percent (namely, the 2, 3, 7, and 10 percent damping ratios), which are not specified in the EPRI report, were generated from the 5 percent damped HRHF horizontal target response spectrum by multiplying the 5 percent damped spectral values by the spectral ratios for the CEUS rock sites given in Table 1 in Appendix C of Standard Review Plan (SRP) 3.7.1 (Reference 6). For spectral frequencies not listed in Table 1, the ratios that were used follow a log-log amplitude-frequency linear interpolation. The HRHF horizontal target response spectrum for the 4 percent damping ratio was generated by interpolating between the spectral values for 3 percent and 5 percent damping ratios on a log scale for the damping ratio and a linear scale for the spectral acceleration.
The HRHF vertical target response spectra for the 2, 3, 4, 7, and 10 percent damping ratios were generated by multiplying the V/H ratios for the CEUS rock site conditions by the corresponding HRHF horizontal target response spectra.

The generated HRHF horizontal and vertical target response spectra for the 2, 3, 4, 5, 7, and 10 percent damping ratios are shown in Chapter 3, Figures 3.7-12 and 3.7-13, respectively. The guidelines and criteria described in SRP 3.7.1, Rev. 4, for Option 1 Approach 1 (Reference 6), were used for generating a set of three-component acceleration time histories compatible with HRHF target response spectra. The generated HRHF horizontal and vertical acceleration time histories, named H1H and H2H for both horizontal directions and VTH for vertical direction, are plotted along with the integrated velocity and displacement time histories that are presented in Chapter 3, Figures 3.7-14, 3.7-15, and 3.7-16, respectively. The comparisons of the time history response spectra with the corresponding HRHF horizontal and vertical target response spectra for each damping value are shown in Chapter 3, Figures 3.7-17, 3.7-18, and 3.7-19.

3.7B.3 High Frequency Site Profiles

Among the eight generic site-shear-wave-velocity profiles (site profiles S1 through S4 and S6 through S9) developed for the APR1400 standard plant design, the site profiles that could be classified as hard-rock sites are S8 and S9. For site profile S8, the depth of bedrock where the rock shear-wave velocity (Vs) is equal to 2,804 m/sec (9,200 ft/sec) is 61 m (200 ft). For site profile S9, the depth of bedrock where the Vs is equal to 2,804 m/sec (9,200 ft/sec) is 30.5 m (100 ft).

Site profile S9 was determined to be more critical when subjected to the HRHF horizontal seismic input motion than site profile S8, based on a comparison of the horizontal site response amplification transfer functions from the bedrock where Vs is equal to 2,084 m/sec (9,200 ft/sec) for site profiles S8 and S9. Therefore, the soil-structure interaction (SSI) analysis using the HRHF seismic input motion was performed for site profile S9. The soil layers, and their associated properties, used in the SSI model for HRHF seismic input motions are shown in Table 3.7B-3.

3.7B.4 Soil-Structure Interaction Model

For the evaluation of the impact of HRHF seismic input motion on the APR1400 standard plant design, the nuclear island SSI model described in Subsection 3.7.2 was analyzed.
using the ACS SASSI computer program. Acceleration time histories compatible with the HRHF target response spectra are applied at the finished grade.

The methodology developed by Tseng and Lilhanand, as described in the 1997 EPRI report TR-102631 (Reference 7), was used for the SSI analysis using ACS SASSI to incorporate spatial incoherence in the seismic input motions. The methodology uses the hard-rock coherency functions for horizontal and vertical seismic ground motions developed by Abrahamson (Reference 8). The analysis considering spatial incoherence of the HRHF seismic input motions was performed for a total of 16 principal coherency modes. The analysis results obtained from SRSS of the responses of modes 1 through 7 and modes 1 through 12 are compared with the corresponding results obtained from SRSS of the responses of modes 1 through 16 (Reference 3). These comparisons indicate that 16 modes are adequate for capturing the incoherent-motion SSI responses of the APR1400 nuclear island structures because the addition of the responses for modes greater than mode 16 leads to insignificant changes in the generated ISRS. Comparisons of in-structure response spectra (ISRS) generated from coherent and incoherent seismic input motions compatible with the HRHF response spectra based on SRSS of 16-mode responses showed little difference in the response spectra below 4 Hz. In the high frequency range above 30 Hz, the incoherent ISRS were lower than the coherent ISRS.

The seismic responses ISRS at selected major locations in nuclear island structures resulting from the CSDRS and HRHF seismic inputs were compared to assess the significance of the HRHF response spectra. The comparison showed that most HRHF response spectra exceed the CSDRS response spectra above 10 Hz. The results are typical of the comparative responses found throughout the APR1400 standard plant.

The exceedances of HRHF-based ISRS in the high frequency range were evaluated to confirm that the high frequency response has marginal effect on the equivalent SSCs qualified by analysis for the ISRS developed from the APR1400 CSDRS.

3.7B.5 Evaluation Methodology

An evaluation of the representative APR1400 SSCs was performed to demonstrate that the APR1400 nuclear power plant is qualified for high frequency seismic response. The evaluation was made on the SSCs that are potentially sensitive to high frequency input selected by screening.
The safety functions of the SSCs selected by screening were assessed using high frequency seismic input to verify that their seismic responses are non-damaging.

3.7B.6 General Selection Screening Criteria

The following general screening criteria were used to identify representative APR1400 SSCs to be evaluated to demonstrate the acceptability of the APR1400 nuclear power plant for high frequency motion:

a. Importance to safety, including the safety function for the safe shutdown earthquake (SSE) event and the potential failure modes due to an SSE. SSCs whose failure modes do not affect the ability to achieve safe shutdown are excluded.

b. Location in areas of the plant that is susceptible to large, high frequency seismic inputs. The analyst verifies that the equipment or structure in the location has high frequency response to seismic input by evaluating the HRHF seismic response spectra at the equipment/structural attachment points.

c. Significant modal response within the region of high frequency amplification, as defined by items such as modal mass, participation factor, stress, and deflection. The analyst determines that the equipment or structure has dynamic response in the region of the seismic input associated with the high frequency amplified response that would result in significant stress or loads that are included in the load combinations.

d. Significant total stress compared to allowable stress in load combinations that include seismic loads. This criterion complements the criterion in item c where it is determined that the seismic stress due to equipment/structure response in the high frequency region is meaningful compared to the allowables and is therefore included in the load combinations.

The portions of the SSCs that were selected for evaluation of their high frequency seismic response were as follows:

a. Building structures
   1) Reactor containment building internal structure
2) Reactor containment building containment structure

3) Auxiliary building

b. Reactor coolant system (RCS)

1) Reactor vessel internals (RVI) and core

2) Component supports and nozzles of the RCS

c. Piping system

d. Safety-related equipment

3.7B.7 Evaluation

3.7B.7.1 Building Structures

Maintaining the structural integrity of the nuclear island structures is important to the safety of the plant. Representative portions of the building structures that were evaluated for the effect of high frequency input were selected based on the areas with the potential to experience high seismic shear and moment loads in a seismic event.

The evaluation consisted of a comparison of the seismic loads and equivalent acceleration from high frequency input to those obtained from the APR1400 design-basis CSDRS for the representative building structures. The nuclear island structures were considered to be qualified for high frequency input if the seismic loads and equivalent acceleration from the CSDRS enveloped those from the high frequency input.

3.7B.7.2 Reactor Coolant System

The RVI support the core and are therefore important to safety. RVI consist of complicated components whose natural frequencies are in the relatively high frequency range.

RCS component supports were selected as one of the evaluation items because they help maintain the capability of RCS components to perform their intended safety-related functions.
Nozzles were evaluated because piping failures generally occur at high stress locations, such as at the nozzles of a component, and they represent the sensitivity of the reactor coolant loop piping to high frequency excitation.

For selected items, the HRHF response was evaluated by comparing the design loads with the loads obtained from the HRHF seismic analyses. It was concluded that RCS supports and nozzles are acceptable for the HRHF seismic loads if the design loads from the CSDRS envelope those from the HRHF seismic analyses.

3.7B.7.2.1 Reactor Vessel Internals and Core

The RVI and core were selected because they are important to safety and their analysis is representative of major primary components. Because the natural frequencies of the RVI components are in the relatively high frequency range, the RVI may be sensitive to high frequency excitation.

A series of analyses were performed to obtain the response of the RVI and core to HRHF loads. The RVI HRHF analysis was done using the HRHF excitation of the reactor vessel (RV) obtained from the response of the reactor containment building and RCS to HRHF loads. Then, the response of the core was calculated using the detailed core model, and the core plate motion was obtained from the RVI analysis.

The time history analyses of the RVI and core were performed for each HRHF mode, and the responses of all modes were combined for the resultant response. The maximum response of each mode was used for the combination. The broadening of the input excitation was also considered for the RVI and core analyses by frequency variation as implemented for CSDRS loads.

3.7B.7.2.2 Component Supports and Nozzles of the RCS

The purpose of RCS structural supports is to support the RCS components during normal operation and transients and during SSE and design basis accident conditions. RCS component supports are necessary to maintain the integrity of the RCS components to preserve their safety function. Support loads at all locations of RCS supports were compared.
The RV is supported by four vertical columns located under the vessel inlet nozzles. The columns are designed to be flexible horizontally to allow for horizontal thermal expansion during heatup and cooldown. The columns also support the RV vertically.

The steam generator (SG) is supported at the bottom by a sliding base bolted to an integrally attached conical skirt. The sliding base rests on low friction bearings that allow unrestrained thermal expansion of the RCS. Two keyways in the sliding base join with embedded keys to guide the movement of the SG during expansion and contraction of the RCS and also limit the movement of the bottom of the SG during SSE and branch line pipe break (BLPB) events.

Reactor coolant pump (RCP) supports consist of four vertical columns, four horizontal columns, and two horizontal snubbers.

The pressurizer (PZR) is supported by a cylindrical skirt that is welded to the pressurizer and bolted to the building structure. Four keys welded to the upper shell of the PZR provide additional restraint for an SSE, pressurizer pilot-operated safety relief valve (POSRV) actuation, and BLPB conditions.

The RCS component nozzles of the RV, SG, and RCP were selected to be included in the evaluation because the nozzle of a component has higher potential for failure than other locations of the components and the cold leg, hot leg, and crossover leg are relatively sensitive to high frequency compared to other components.

3.7B.7.3 Piping System

Because piping lines and piping supports throughout the plant are designed according to guidelines, the stress analysis of a sample of lines is representative of all lines in the plant. Susceptibility to excitation caused by high frequency input depends on the following factors:

- a. The local HRHF ISRS have exceedances in the high frequency range relative to the APR1400 CSDRS ISRS.

- b. The piping system has modes or natural frequencies in the high frequency range.
c. The piping system layout includes valves or other concentrated masses that have closely spaced supports and therefore cause high local natural frequencies. The layout generally yields significant cumulative mass in the high frequency range.

American Society of Mechanical Engineers (ASME), Class 1, 2, and 3 piping systems are required to be evaluated for the HRHF seismic response spectra.

A graded approach is taken to the scope of piping systems and components design. Therefore, the piping systems within the scope of the graded approach are evaluated for the HRHF seismic response spectra.

The combined license (COL) applicant is to evaluate the HRHF response spectra for piping systems other than those within the scope of the graded approach (COL 3.7B(3)).

3.7B.7.4 Safety-Related Equipment

As a result of the high frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. The use of prior testing results should be justified by demonstrating that the frequency content of the PSD of the test waveform is sufficient.

Safety-related equipment is evaluated for the effect of high frequency input motion to demonstrate their safety-related functionality.

For the evaluation of the equipment and components functionality for those cases where the GMRS/FIRS-based ISRS exceed the CSDRS-based ISRS below 50 Hz, further equipment and component functionality evaluations are needed. The screening process is applied for identification and evaluation of high-frequency sensitive mechanical and electrical equipment/components. For the new qualification test of equipment and components, the RRS that is generated to meet GMRS/FIRS-based ISRS as well as the CSDRS-based ISRS are applied.

Evaluation process for evaluating equipment and components that are screened in is described in Technical Report APR1400-E-S-NR-14004-P (Reference 3) with a basis for the criteria used for each screening step that is used to identify equipment/components with potential to HF sensitivity.

The COL applicant is to perform the HRHF evaluation of safety-related equipment. (COL 3.7B(1)). The seismic qualification test/analysis will be performed for the components to
envelop the in-structure response spectra resulting from the entire set of certified seismic design response spectra (CSDRS), including ground motions for the COL sites with high frequency content.

3.7B.7.4.1 Evaluation process steps and description

Identification and evaluation process of HF sensitive mechanical and electrical equipment and components is performed for safety-related equipment and components before performing seismic qualification.

3.7B.7.4.1.1 Potentially high-frequency sensitive equipment

Safety-related equipment and components that have been undergone prior qualification testing/analysis are classified to either HF sensitive or HF insensitive. The potentially HF sensitive equipment and components are evaluated for the HF sensitivity.

The concern with potentially HF-sensitive components is related to the functionality of the devices when subjected to HF motions.

The component types suggested in Table 3.7B-3 are considered potentially sensitive to HF motions and should be screened based on the procedures and criteria provided in Technical Report APR1400-E-S-NR-14004-P (Reference 3). For all safety-related equipment that is designed to have part(s) of its assembly with components classified to be potentially sensitive to HF motions are to be evaluated for the adequacy in accordance with the procedures described in APR1400-E-S-NR-14004-P (Reference 3).

List of HF sensitive equipment is provided to Technical Report APR1400-E-X-NR-14001-P, “Equipment Qualification Program” (Reference 13), Table 3, “Equipment Qualification Equipment List.” The COL applicant is to verify the applicability of evaluation of the items potential to HF sensitivity (COL 3.7B(2)).

3.7B.7.4.1.2 Items not potentially HF sensitive

In case that items are not potentially sensitive to HF, the confirmation that the natural frequency of the equipment or components is at the region where CSDRS-based ISRS exceedance occurs is required. When the natural frequency is at the CSDRS exceedance region, the higher seismic load based on the HRHF-based ISRS is imposed to the equipment compared to the CSDRS-based ISRS. Therefore additional evaluation to ensure
that HRHF-based ISRS does not affect any structural integrity and functional requirement is required although the equipment is potentially classified to be insensitive to HF. The method of additional evaluation could be evaluating the existing data to envelop the RRS that is prepared for HRHF-based ISRS including margins HRHF-based ISRS, conduct screening test if necessary or any analysis to verify acceptability.

3.7B.8 Combined License Information

COL 3.7B(1) The COL applicant is to perform the HRHF evaluation of safety-related equipment.

COL 3.7B(2) The COL applicant is to verify the applicability of evaluation of the items potential to HF sensitivity.

COL 3.7B(3) The COL applicant is to evaluate the HRHF response spectra for piping systems other than those within the scope of the graded approach.

3.7B.9 References


### 5%-Damped HRHF Horizontal Target Response Spectrum

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(1) Unit weight density of soil/rock

(2) Poisson’s Ratio
### Potentially HF Sensitive Component

- Electro-mechanical relays (e.g., control relays, time delay relays, protective relays)
- Electro-mechanical contactors (e.g., Motor Control Center (MCC) starter)
- Circuit breakers (e.g., molded case and power breaker - low and medium voltage)
- Auxiliary contacts (e.g., for Molded Case Circuit Breaker (MCCBs), fused disconnects, contactors/starters)
- Control switches (e.g., benchboard panel, operator switches)
- Transfer switches (e.g., low and medium voltage switches with instrumentation)
- Process switches and sensors (e.g., pressure/diff. pressure, temperature, level, limit/position, and flow)
- Potentiometers
- Digital solid-state devices (mounting and connections only)
- Microprocessors-based components
- Connectors and connections (including circuit board connections for digital and analog equipment)
- Unrestrained components
Figure 3.7B-1 Comparison of 5%-Damped HRHF and CSDRS Horizontal Target Response Spectra
Figure 3.7B-2  Comparison of 5%-Damped HRHF and CSDRS Vertical Target Response Spectra
Figure 3.7B-3  5%-Damped HRHF Horizontal Target Response Spectrum
Figure 3.7B-4  5%-Damped HRHF Vertical Target Response Spectrum
3.8 Design of Category I Structures

3.8.1 Concrete Containment

3.8.1.1 Description of the Containment

3.8.1.1.1 Basic Configuration

The containment encloses the reactor vessel, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safety features systems. The containment provides reasonable assurance that leakage of radioactive material to the environment does not exceed the acceptable dose limit as defined in 10 CFR 50.34 (Reference 1) even if a loss-of-coolant accident (LOCA) occurred.

The internal structures are physically independent of the containment, except at the supporting foundation basemat. The connections of operating and intermediate floors to the containment wall are described in Subsection 3.8.3.1.10.

The containment shares a common basemat with the auxiliary building. The auxiliary building wraps around the containment with a seismic isolation gap of 150 mm (6 in.).

The reactor containment building basemat has a continuous tendon gallery that provides access to the vertical tendons below the wall-basemat junction.

The containment is a prestressed concrete structure composed of a right circular cylinder with a hemispherical dome and is founded on safety-related common basemat. The structures are lined on the inside with steel plate that acts as a leak-tight membrane. The cylindrical portion of the containment is prestressed by a post-tensioning system consisting of horizontal and inverted “U” vertical tendons. There are three buttresses equally spaced around the containment wall, and each horizontal tendon is anchored at buttresses 240 degrees apart, bypassing the intermediate buttress. The dome portion is prestressed by a post-tensioning system consisting of horizontal tendons up to a 45-degree vertical angle and of two groups of inverted “U” vertical tendons oriented 90 degrees to each other. The inverted “U” tendons are carried through the cylindrical wall and anchored at the tendon gallery.

The general configuration and dimensions of the containment structure are shown in Figures 3.8-1 and 3.8-2. Local areas at the equipment hatch and two personnel airlock areas are thickened as shown in Figure 3.8-3.
The containment has the following dimensions:

a. Inside diameter of containment: 45.72 m (150 ft)

b. Inside height of containment: 76.66 m (251.5 ft) from the top of base slab to the ceiling of dome apex

c. Thickness of containment wall: 1.37 m (4 ft 6 in.)

d. Dome thickness: 1.22 m (4 ft)

3.8.1.1.2 Foundation Basemat

A description of the foundation basemat is given in Subsection 3.8.5.

Appendix 3.8A shows the reinforcement details of the intersection where the containment wall intersects with nuclear island (NI) common foundation basemat.

The design basis of the NI common basemat under the RCB and the AB conform to the requirement of ASME Section III, Division 2, CC and ACI 349, respectively. The boundary of jurisdiction between the ASME and the ACI codes is shown in Figure 3.8-26. The jurisdictional boundary is placed at the thickness transition interface due to the physical configuration and functional requirements of the basemat. The anchoring of the containment shell reinforcement is limited within the ASME Code boundary as shown in Figures 3.8A-16 and 3.8A-17.

3.8.1.1.3 Containment Shell

3.8.1.1.3.1 General

The cylindrical containment shell has a constant thickness of 1.37 m (4 ft 6 in.) from the top of the foundation basemat to the springline. The shell is thickened locally around the equipment hatch, two personnel airlocks, feedwater, and main steam line penetrations. The containment reinforcing consists primarily of hoop and meridional steel. Prestressing tendons are also arranged in hoop and meridional directions. The tendons in the meridional directions extend over the dome to form an inverted “U” shape.

Structural steel supports are provided for distribution systems on the containment shell and dome, including the electrical conduit, cable tray, and spray system piping. The supports
are welded to embedment plates on the inside and outside of the containment shell and dome.

3.8.1.1.3.2 Reinforcing Bar Layout

Continuous hoop and meridional reinforcements are placed at the outside layers of the cylindrical wall. Similar reinforcements are also provided at the inside layers. Additional reinforcing bars are provided around major penetrations in the shell as required. Shear ties are provided where radial shear reinforcing is required.

3.8.1.1.3.3 Prestressing Tendon Layout

The cylindrical wall is post-tensioned with 165 hoop tendons at 305 mm (12 in.) center. Each hoop tendon is anchored at the buttresses located 240 degrees apart, bypassing the intermediate buttress. Thus, three hoop tendons make two complete rings. Vertical post-tensioning for the cylindrical wall is provided by two sets of 50 equally spaced orthogonal inverted “U” vertical tendons.

The spacing of the hoop tendons and inverted “U” vertical tendons in the cylindrical wall is shown in Figure 3.8-4.

3.8.1.1.3.4 Liner Plate Details and Anchorage

The 6.0 mm (0.25 in.) liner plate is attached to inside of the cylindrical wall by vertical angles embedded in the cylindrical wall as shown in Figure 3.8-5. Horizontal stiffeners are provided for the liner plate to serve as an inside formwork for the cylindrical wall during concrete placement. The details for the containment liner below the CIS and anchorage are shown in Figures 3.8-6 and 3.8-7.

3.8.1.1.3.5 Penetrations

Access to the interior of the containment is provided through two personnel airlocks. An equipment hatch permits transfer of equipment into and out of the containment. In addition to these access openings, other major penetrations provided in the containment wall are for the main steam, feedwater, and HVAC lines. The containment wall is also penetrated by various process pipelines (Figure 3.8-8), electrical penetration assemblies (Figure 3.8-9), and the fuel transfer tube penetration. Descriptions are provided in Subsections 3.8.2.1.3.1.1 and 3.8.2.1.3.1.2. The penetrations greater than 457.2 mm (18 inches), except the personnel airlocks and equipment hatch, are given in Table 3.8-9.
The penetration sleeves through the containment are fabricated of steel, anchored to the concrete structure, and seal welded to the containment liner plate. The type, size, and location of the penetration, as well as any load that is imposed on the cylindrical wall by the penetration, determine whether any additional reinforcing is required in the wall around the penetration. Post-tensioning tendons are deflected around the penetrations, if required. Minimum bend radius of all deflected tendons is 9.15 m (30.0 ft) between tangent points. Portions of the containment pressure boundary that are steel and not backed by concrete, such as the equipment hatch, personnel airlocks, and Class MC penetration assemblies including the fuel transfer tube penetration sleeve, are designed in accordance with ASME Section III, Division 1, NE (Reference 2), as described in Subsection 3.8.2.

3.8.1.1.3.5.1 Personnel Airlocks

Access to the containment is provided through two personnel airlocks with a diameter of 3.4 m (11 ft 2 in.). The personnel airlocks are located at the plant grade level (Az. 280 degrees and elevation 103 ft 9 in.) and the operating floor level (Az. 234 degrees and elevation 159 ft 9 in.). The containment wall around the personnel airlocks is thickened, and additional reinforcement is provided for stress concentrations due to the opening, as shown in Figure 3.8-3 and Figure 3.8A-10. The post-tensioning tendons are deflected around the penetrations.

3.8.1.1.3.5.2 Equipment Hatch

The equipment hatch, with a diameter of 7.92 m (26 ft), is located at Az. 280 degrees and elevation 167 ft 6 in. The containment wall around the equipment hatch is thickened, and additional reinforcement is for stress concentrations due to the opening provided around the opening. The post-tensioning tendons are deflected around the opening.

3.8.1.1.3.6 Polar Crane Bracket

The polar crane brackets are spaced equally and embedded in the cylindrical wall to support the polar crane girder. Forces acting on the wall due to bracket loads are accommodated by additional reinforcement in the wall. A thickened liner plate is provided in the bracket areas. The spacing of the vertical liner stiffener in the vicinity of thickened liner plates is designed to limit excessive strain in the adjoining 6.0 mm (0.25 in.) liner plate.
3.8.1.1.4 Containment Dome

3.8.1.1.4.1 General

The roof of the containment is a hemispherical dome. The inside of the dome is lined with a steel liner plate to provide leak-tightness. The buttresses are extended up to 48 degrees into the dome to provide anchorage for the dome hoop tendons.

3.8.1.1.4.2 Reinforcing Bar Layout

The containment dome is reinforced using the same arrangements as the dome tendons. Orthogonal reinforcing is the continuation of the vertical reinforcing in the cylindrical wall. Hoop reinforcing is also provided up to 45 degrees above the springline to complete the system. Radial ties are also provided over the entire dome to account for radial tension due to the curvature of the prestressing tendons.

3.8.1.1.4.3 Prestressing Tendons

Post-tensioning for the dome is provided by the orthogonal inverted “U” vertical tendons and 30 hoop tendons at 1.5 degrees on the center up to 45 degrees above the springline. The spacing of hoop tendons in the dome is shown in Figure 3.8-4.

3.8.1.1.4.4 Liner Plate Details and Anchorage

Radial and hoop stiffeners are provided for attaching the 6.0 mm (1/4 in.) liner plate to the concrete dome, as shown in Figure 3.8-24. However, in the central portion, stiffeners are placed to form a rectangular grid. The liner serves as inside formwork for placing the concrete.

3.8.1.2 Applicable Codes, Standards, and Specifications

The following regulations, codes, standards, and specifications are used in the design of the concrete containment.

3.8.1.2.1 Design Codes, Standards, Specifications, and Regulations

The design codes, standards, specifications, and regulations are listed in Table 3.8-1. The primary design code for concrete containment is ASME Section III, Division 2, CC (Reference 3).
3.8.1.2.2 NRC Regulatory Guides

Conformance to each NRC Regulatory Guide (RG) is described in Section 1.9. The NRC RGs applicable to the design of the concrete containment are NRC RG 1.35.1 (Reference 5), NRC RG 1.136 (Reference 6), NRC RG 1.91 (Reference 30), NRC RG 1.216 (Reference 50), NRC RG 1.221 (Reference 46), and NRC RG 1.7 (Reference 7).

3.8.1.2.3 Industry Standards

Internationally recognized industry standards published by ASTM are used whenever possible to define material properties, testing procedures, and fabrication and construction methods.

3.8.1.3 Loads and Load Combinations

The containment is designed to resist the loads given in CC-3000 of the ASME Code and NRC RG 1.136 with the exceptions listed below.

a. The post-LOCA flooding combined with the safe shutdown earthquake (SSE) is more severe than the post-LOCA flooding combined with the operating basis earthquake (OBE) set at one third or less of the SSE for the plant. Therefore, only the post-LOCA flooding SSE combination is considered in the design.

b. CC-3720 of the ASME Code is satisfied when the containment structure is exposed to the load combination listed below. As a minimum design condition, the pressure \((P_{g1} + P_{g2})\) is not less than 310 kPa (45 psig).

\[
D + F + P_{g1} + P_{g2}
\]

Where:

\[
\begin{align*}
D &= \text{Dead load} \\
F &= \text{Prestress} \\
P_{g1} &= \text{Pressure resulting from an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction} \\
P_{g2} &= \text{Pressure resulting from uncontrolled hydrogen burning}
\end{align*}
\]
A description of load categories and definition of loads are given in Subsections 3.8.1.3.1 and 3.8.1.3.2.

3.8.1.3.1 Load Category

The load categories include any condition encountered during construction and testing, and in the normal operation of a nuclear power plant, as well as the conditions resulting from extreme environmental conditions postulated during the life of the facility and certain combinations thereof.

The design loads are defined as service load category and factored load category depending on the frequency of their occurrence.

3.8.1.3.1.1 Service Loads

Service loads are any loads encountered during construction and in the normal operation of a nuclear power plant and include loads such as any anticipated transient or test loads during normal and emergency startup and shutdown of the nuclear steam supply, safety and auxiliary systems, and the severe environmental loads that may be anticipated during the life of the facility.

Construction

The construction condition considers events and loads during construction, including the various stages of prestressing but excluding those during testing. Contraction loads for buildings and other structures are developed in accordance with Table 3.8-2 and with SEI/ASCE 37-02 (Reference 8).

Testing

The testing condition includes a consideration of events and loads applied during the containment structural integrity or leak rate testing and preoperational tests such as hydrostatic testing of equipment. Each testing event is considered to be mutually exclusive of other testing events.

Normal

The normal condition includes a consideration of events and loads that are reasonably expected during the operation, shutdown, and normal maintenance of the power plant.
Factored loads include loads encountered infrequently, such as severe environmental, extreme environmental and abnormal loads.

**Severe Environmental**

The severe environmental condition includes a consideration of the loads due to infrequent site-related environmental events such as operating basis earthquake, design basis wind, and design basis flood or precipitation during the plant life.

**Extreme Environmental**

The extreme environmental condition includes a consideration of the loads due to site-related environmental events that are credible but highly improbable. These events include the SSE, design basis tornado/hurricane and its associated missiles, probable maximum precipitation, probable maximum water level, and the site-related accidents to be postulated but not included in the abnormal loading category.

**Abnormal**

The abnormal condition includes a consideration of the loads due to design basis events. They include pressure, temperature, blast, pipe whip, jet impingement, flooding, and pipe reactions due to postulated pipe breaks for design basis accidents. This loading condition also includes plant-related non-environmental missiles. The loads from each postulated accident event are considered to be mutually exclusive of other postulated accidents.

**Abnormal/Severe Environmental**

The abnormal/extreme condition includes a consideration of the loads due to the highly improbable simultaneous occurrence of abnormal and severe environmental loading conditions. Only the specified combinations of these conditions are considered.

**Abnormal/Extreme Environmental**

The abnormal/extreme condition includes a consideration of the loads due to the extremely improbable simultaneous occurrence of abnormal and extreme environmental loading conditions. Only the specified combinations of these conditions are considered.
3.8.1.3.2 **Design Loads**

The design loads pertaining to the design of containment are as follows:

a. **Dead load (D)**

   Dead loads, including hydrostatic and permanent equipment loads

b. **Live load (L)**

   Live loads, including any movable equipment loads and other loads that vary with intensity during each occurrence such as soil pressures

c. **Prestress (F)**

   Loads resulting from the application of prestress, including effects resulting from the construction sequence used to post-tension the tendon

d. **Operating temperature (T_o)**

   Thermal effects and loads during normal operating conditions, based on the most critical transient or steady-state condition; the combination of internal and external temperatures that produce the maximum effects is considered.

e. **Pipe reaction (R_o)**

   Pipe reaction during normal operating or shutdown conditions, based on the most transient or steady-state condition

f. **External pressure (P_v)**

   Pressure loads resulting from pressure variation either inside or outside the containment

g. **Test pressure (P_t)**

   Pressure during the structural integrity and leak rate tests

h. **Test temperature (T_t)**

   Thermal effects and loads during the structural integrity and leak rate tests
i. Wind load \((W)\)

Loads generated by the design wind specified for the plant site

j. Seismic load \((E_o)\)

OBE loads are not applicable to the containment design for the APR1400 because an OBE level is one third of the SSE.

k. Seismic load \((E_s)\)

Loads generated by the SSE; only the actual dead loads and, 25 percent of live load or 75 percent of snow are considered in evaluating seismic response forces.

l. Tornado or hurricane load \((W_t)\)

Tornado or hurricane loading including the effects of missile impact

m. Internal flooding \((H_a)\)

Load resulting from internal flooding other than from pipe breaks

n. Accident pressure \((P_a)\)

Design pressure load within the containment generated by the design basis accident, based on the calculated peak pressure with an appropriate margin

o. Accident temperature \((T_a)\)

Thermal effects and loads generated by the design basis accident including operating temperature \((T_o)\)

p. Pipe reaction \((R_a)\)

Pipe reaction from thermal conditions generated by the design basis accident including pipe reaction at normal operating or shutdown conditions \((R_o)\)

q. Pipe break load \((R_r)\)
Local effects due to the design basis accident normally include all postulated high-energy system ruptures. These loads include an appropriate dynamic load factor to account for the dynamic nature of the load. This load category includes:

1) Pipe break reaction load ($Y_r$)

$Y_r$ is defined as the equivalent static load on the structure generated by the reaction of the high-energy pipe during the postulated break.

2) Pipe break jet impingement load ($Y_j$)

$Y_j$ is defined as the jet impingement equivalent static load on the structure generated by the postulated break.

3) Pipe break missile impact loads ($Y_m$)

$Y_m$ is defined as the missile impact equivalent static load on the structure generated by or during the postulated break, such as pipe whipping.

r. Flooding load ($Y_f$)

$Y_f$ is the load within or across a compartment or building due to flooding generated by a postulated pipe break. These loads are calculated considering the design basis flood heights.

s. Other Loads

Other loads refer to postulated events or conditions that are not included in the design basis. However, the safety of the containment under combustible gas loads and aircraft impact are assessed and demonstrated to conform with the allowable values in CC-3720 of the ASME Code and NEI 07-13, respectively. This load category includes:

1) Aircraft hazard (A)

Aircraft hazard refers to loads on a structure resulting from the impact of an aircraft. The evaluation of this loading condition is considered as part of the plant safeguards and security measures.
2) Combustible gas load ($P_g$)

Combustible gas loads are pressure loads that result from a fuel-clad metal-water reaction ($P_{g1}$) followed by an uncontrolled hydrogen burn ($P_{g2}$) during a post-accident condition in the containment inerted by carbon dioxide. NRC RG 1.136, Regulatory Position C.5 provides the loads and load combinations acceptable for analysis and design of containment when exposed to the loading conditions associated with combustible gas. The loads and load combinations for combustible gas are provided in Subsection 3.8.1.3.

t. Missile loads other than hurricane generated or tornado-generated missiles

There are no missile loads on the containment resulting from activities of nearby military installations, turbine failures, or other causes.

u. Valve actuation load ($G$)

Loads resulting from relief valve or other high energy device actuation

v. Design flood/precipitation load ($H$)

Flood loads on seismic Category I structures are determined based on the maximum site flood levels specified in Chapter 2.

w. Probable maximum flood/precipitation ($H_s$)

$H_s$ is the forces, due to the probable maximum precipitation as well as the maximum flood level, which includes the effects of seiches, surges, waves, and tsunamis.

x. Crane and trolley loads ($C$)

This load is the crane and trolley lifted load, including impact load, longitudinal load, and lateral load. All of these loads shall be considered as acting simultaneously. This load is detailed in Subsection 3.8.4.3.1. In the case of the crane lifting load, it is not considered under severe environmental, extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental loading condition since the polar crane is not permitted to lift any loads during plant operation. When not in use, the polar crane is required to be in the parking position (location of polar crane: Az.280°, trolley
location: 12 ft 7 in away from end of east part), since the crane is placed in the parking position during plant operation. The self-weight for all load conditions and live load (for construction and normal load condition) of polar crane are applied to the parking position (COL 3.8(21)).

3.8.1.3.3 Design Load Combinations

The applicable load combinations and load factors for the design of a concrete containment conform with the requirements of CC-3000 of the ASME Section III, Division 2. Table 3.8-2 lists the load combinations used in the design of the containment.

3.8.1.3.4 Liner Plate Loads and Load Combinations

The load combinations shown in Table CC-3230-1 of the ASME Code are applicable to the liner, except that load factors for all load cases are taken as equal to 1.0. Strains associated with construction-related liner deformations are excluded when calculating liner strains for the service and factored load combinations.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General

The design and analysis procedures are in conformance with the requirements of CC-3000 of the ASME Section III, Division 2.

Throughout the analysis special attention is given to, the following areas of the containment:

a. Intersection between the basemat and the cylinder

b. Intersection between the cylinder and the dome

c. Areas around large penetrations

d. Areas around polar crane brackets

e. Behavior of the base slab relative to the underlying foundation material

f. Stresses due to transient temperature in the liner plate and concrete

g. Penetrations and points of concentrated loads
h. Buttresses

A typical section of the reactor containment building is shown in Figures 3.8-10 and 3.8-11. The computer programs utilized in the analysis and design of seismic Category I structures are described in Table 3.8-11 and Table 3.8A-40.

3.8.1.4.2 Containment Structure

The ANSYS (Reference 9) computer program is used to analyze the containment for the loads defined in Subsection 3.8.1.3.2. The analysis results of these load case analyses are combined and factored using the loading combinations defined in Subsection 3.8.1.3.3.

The analysis model of the containment consists of the dome, cylindrical wall, and a part of the basemat. No other structures are physically connected to the containment structure; therefore, the basemat is the only interfacing part in the containment model. Subsection 3.8.5 describes the modeling of the common basemat structure.

A three-dimensional, eight-node solid element that is suitable for moderately thick shell structures is used to model the containment concrete dome and cylindrical wall. Five layers of solid elements through the thickness are used to model the dome and cylindrical wall. The buttresses and thickened areas around the large penetrations are included in the ANSYS model. The structural analysis model of containment shell and dome is shown in Figure 3.8-12.

Post-tensioning tendon forces are included in the containment structural analysis. Forces for each tendon in the cylindrical wall and dome are averaged along their routing lengths in the containment model. Post-tensioning forces are transferred to the concrete by imposing initial stresses along the lengths of the modeled tendons which are embedded in the concrete model. The tendons are modeled using two-node three dimensional type elements. The prestress losses are explicitly included in the initial stress values.

Additional descriptions of the containment computer model are provided in Appendix 3.8A.

3.8.1.4.3 Assumptions on Boundary Conditions

The boundary condition is applied at the bottom of the basemat and at vertically cut faces inside and outside the basemat. The cut faces are fixed radially and circumferentially,
whereas the bottom face is fixed only vertically so that unimportant parts can be cut and removed.

3.8.1.4.4 Axisymmetric and Non-Axisymmetric Loads

The containment is modeled in its entirety as a three-dimensional structure. The loads described in Subsection 3.8.1.3.2 are applied in the locations and directions appropriate for each load. Overall pressure is applied uniformly to the interior surface of the containment structure.

Localized loads, such as penetration dead loads, hydrostatic pool water loads, live loads, and pipe rupture loads are applied to specific portions of the structural model as appropriate. Post-tension loads are applied to each tendon in its specific location. Localized loads at buttress locations that are caused by the offset of the tendons with respect to the shell are accounted for in the modeling. Response spectrum analysis is applied for the two horizontal and one vertical seismic load directions using the methodology described in Subsection 3.7.2.

3.8.1.4.5 Transient and Localized Loads

3.8.1.4.5.1 Analysis of Areas of Containment Wall Supporting Polar Crane Brackets

The containment wall around the crane brackets is analyzed considering the effects of crane bracket reactions. To account for potential difference in the timing between containment construction and polar crane installation, two models are used for the analyses: the overall containment full model and a partial model with only the containment cylinder.

3.8.1.4.5.2 Thermal Stress Analysis

To analyze the containment for thermal gradients, the nonlinear temperature profile across the containment wall thickness is obtained through the transient heat analysis using the ANSYS program. The resultant forces and moments from the thermal stresses are applied to each design section along with other appropriate axial forces and moments due to mechanical loads acting simultaneously with the thermal loads.

The stresses in the concrete and reinforcing steel are checked by using the DARTEM program, which is verified and validated, taking into account the self-limiting effects of thermal moments due to concrete cracking.
3.8.1.4.6 Creep and Shrinkage Analysis

The effects of concrete creep, shrinkage, elastic shortening, and tendon steel relaxation are included in the computations for prestress losses in the tendons.

a. Concrete creep strain

1) Vertical direction = \(592 \times 10^{-6}\) mm/mm (in/in)
2) Horizontal direction = \(930 \times 10^{-6}\) mm/mm (in/in)

b. Concrete shrinkage strain = \(120 \times 10^{-6}\) mm/mm (in/in)

c. Poisson’s ratio = 0.17

d. Elastic modulus

1) Containment external concrete wall = 30,441.74 MPa (4.415 \times 10^6\) psi
2) Containment internal concrete structure = 30,441.74 MPa (4.415 \times 10^6\) psi
3) Containment concrete basemat = 27,789.38 MPa (4.031 \times 10^6\) psi
4) Prestressing steel material = 193,053.20 MPa (2.8 \times 10^7\) psi

e. Elastic shortening of concrete

1) Vertical direction = \(124 \times 10^{-6}\) mm/mm (in/in)
2) Horizontal direction = \(194 \times 10^{-6}\) mm/mm (in/in)

f. Tendon relaxation = 6%

g. Friction coefficients

There are two types of friction losses, wobble friction loss from unintended curvature and curvature friction loss from intended curvature. The two kinds of curvature in tendons induce friction loss of tensile stress. Friction losses are calculated in accordance with CC-3542 of the ASME Code and taken into account in the design of the post-tensioning system. The friction coefficients are determined experimentally and verified by testing while stressing tendons.
3.8.1.4.7  Tangential Shear

The design and analysis procedures for tangential shear are in accordance with ASME Section III, Division 2 and NRC RG 1.136.

Tangential shear is resisted by the vertical reinforcement and the horizontal hoop reinforcement in the containment wall.

3.8.1.4.8  Variations in Physical Properties

In the design and analysis of the containment, consideration is given to the effects of possible variations in the physical properties of materials on the analytical results. The properties used for analysis purposes were established based on engineering experience with similar construction and materials gained through the construction of Shin-Kori units 3&4 in Korea. The values that were used are delineated in Subsection 3.8.1.4.6. The values are to be evaluated and confirmed by certified material test reports (CMTRs) and concrete long-term material testing. The COL applicant is to perform concrete long-term material testing in a way which verifies physical properties of materials used during the design stage and the characteristics of long term deformation of concrete (COL 3.8(1)). The test results are to be reviewed by designers. Additional reviews of materials and their effects on the analysis and design of the containment will be included in design specification development and materials selection.

Losses due to elastic shortening, concrete creep and shrinkage, and relaxation of the post-tensioning cables were accounted for in the analysis. Subsection 3.8.1.4.6 summarizes parameters to consider time-dependent losses such as shrinkage, creep, and tendon relaxations.

When designing the structure under service and factored load conditions, allowable stress levels are used based on the minimum strength of the concrete and reinforcing materials used in construction of the containment to account for variations in physical properties.

3.8.1.4.9  Analysis of Areas around Large Penetrations

Large penetrations can cause stress concentrations. Increased thickness of the containment shell around these areas preserves the overall shell stiffness, compensates for the effects of stress and strain concentrations, and provides space for post-tensioning tendons and reinforcing bars.
The local structural behaviors at large, thickened penetration regions are investigated using the three-dimensional finite element full model.

3.8.1.4.10 Steel Liner Plate and Anchors

Design of the liner plate includes a consideration of construction loads, internal pressure, prestress, creep, shrinkage, and thermal effects. The design of the containment liner conforms with the allowable stress and strain values in CC-3720 of Section III of the ASME Code.

The liner plate is not assumed as a structural member in the design other than during construction. Loads caused by thermal growth of the liner plate are considered in the concrete containment design.

The liner plate is anchored to the concrete so that the liner stain does not exceed the allowable strain given in the code. The anchorage system is designed so that progressive failure of the anchors or massive bucking of the liner does not occur.

The liner anchorage system is analyzed by using the LBAP program, which is verified and validated, calculating the force and deflection at anchorage points. The design of the liner anchorage conforms with the force and displacement allowables in CC-3730 of Section III of the ASME Code.

For the structural design of containment liner plates, the stresses at formworks are calculated for basemat liner, shell liner, and dome liner, respectively. The lowest ratio of allowable stress to induced stress for each part is shown in Table 3.8-10 as margins of safety for the design.

3.8.1.4.11 Ultimate Pressure Capacity

The ultimate pressure capacity (UPC) of the prestressed concrete containment, which is for assessment of the safety margin above the design-basis accident pressure, is evaluated based on the design results (rebar arrangements) of the structure. A full three-dimensional finite element model is developed for the analysis of the concrete containment. Material nonlinear models for steel and concrete are constructed on the basis of the design code and a few references. For simulating the cracking behavior of concrete, smeared crack model is adopted and the tension stiffening effect and their interaction are also taken into consideration. The reinforcement in concrete structures is provided by means of rebars,
with this modeling approach, the concrete behavior is considered independently of the rebars. Therefore, in the concrete modeling, the tension stiffening is required in the smeared crack model to simulate load transfer across cracks through the rebar which consider the effects of the reinforcement interaction with concrete. The steel is assumed to be a linear elasto-plastic model. The stress-strain curves for the reinforcing steel and tendons are based on the ASME code-specified minimum yield strengths. An elastic-plastic and a piece-wise linear stress-strain relationship above yield stress is used for the reinforcing steel and tendons.

In the initial state of the nonlinear analysis, the containment structure is subject to dead and prestressing loads. During the UPC analysis, the internal pressure is monotonically increased until a specified failure criterion is reached. The pressure corresponding to failure criterion of the liner, rebar, and tendons is recorded. The pressure at which the first failure criterion is reached is determined to be the ultimate pressure capacity of the prestressed concrete containment.

The design and analysis procedures for determining the UPC are performed in accordance with NRC RG 1.216, and is estimated based on satisfying both of the following strain limits: (1) a total tensile average strain in tendons away from discontinuities of 0.8 percent, which includes the strains in the tendons before pressurization (typically about 0.4 percent) and the additional straining from pressurization; and (2) a global free-field strain for the other materials that contribute to resist the internal pressure (i.e., liner, if considered, and rebars) of 0.4 percent. In addition, the additional failure modes, such as concrete shear and concrete crushing which may occur near discontinuities should be considered.

The ultimate pressure capacity of the containment is a pressure of 1.089 MPa (158 psi), at which the maximum strain of the liner plate is approximately 0.4 percent. It is noted that this UPC pressure is the lowest pressure from the acceptance criteria in NRC RG 1.216, and is determined to occur near the upper portion of the equipment hatch. At this ultimate pressure level, the maximum strains of the rebars and tendons do not reach the allowable limit strain values. In addition, with regard to the punching shear (local failure of concrete) at the near discontinuities such as equipment hatch, the shear capacity of shear rebars exceed the shear force corresponding to the ultimate pressure level. However, the concrete shear strength is neglected.
The COL applicant is to provide the detailed design results and evaluation of the ultimate pressure capacity of penetrations, including the equipment hatch, personnel airlocks, electrical and piping penetrations in accordance with NRC RG 1.216 (COL 3.8(2)).

3.8.1.4.12 Combustible Gas Control Inside Containment

The safety of containment under the combustible gas load \( (P_g) \) condition, which includes hydrogen generation pressure load due to 100 percent fuel-clad and water interaction \( (P_g) \) accompanied by hydrogen burning \( (P_g) \), is assessed and demonstrated to comply with the allowable values in ASME Section III, CC-3720. In NRC RG 1.216, for concrete containments, the acceptance criteria are limited to demonstrating that the liner strains satisfy the Factored Load Category requirements presented in ASME Section III, Division 2, CC-3720.

Under these conditions, the loadings should not produce strains in the containment liner plate in excess of the limits established in ASME Section III, CC-3720. Allowable strains for factored loads considering membrane only are 0.005 cm/cm in compression and 0.003 cm/cm in tension. Allowable strains for factored loads considering combined membrane and bending are 0.014 cm/cm in compression and 0.010 cm/cm in tension.

The three-dimensional finite element (FE) model for safety evaluation during the combustible gas load condition is based on the structural analysis model for section design. The FE program is used for the nonlinear analysis of the containment structure. The full FE model includes the entire prestressed concrete containment structure which consists of the concrete wall and dome, the liner plate, rebars, and tendons. The solid and shell elements are used for concrete and liner plate, respectively. In addition, the rebars and tendons are modeled as truss elements. Material nonlinear models for steel and concrete are constructed on the basis of the design code and a few references. For simulating the cracking behavior of concrete, smeared crack model is adopted and the tension stiffening effect and their interaction are also taken into consideration. The reinforcement in concrete structures is provided by means of rebars, with this modeling approach, the concrete behavior is considered independently of the rebars. Therefore, in the concrete modeling, the tension stiffening is required in the smeared crack model to simulate load transfer across cracks through the rebar which consider the effects of the reinforcement interaction with concrete. The steel is assumed to be a linear elasto-plastic model. The stress-strain curves for the reinforcing steel and tendons are based on the ASME code-

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specified minimum yield strengths. An elastic-plastic and a piece-wise linear stress-strain relationship above yield stress is used for the reinforcing steel and tendons.

For the structural analysis under the combustible gas load, the dead load is applied first and then the pressure load is incremented until the pressure from this event is reached. Since the duration of the loading is much larger than the period of the structure, the static analysis methods are adequate for structural integrity evaluation under the combustible gas load. Based on the results of the analyses, all of the tendons and rebars are still in the elastic stage. At the maximum pressure loading level of the combustible gas load condition, the liner plate strains at the cylindrical wall base, mid-height wall, and penetration regions do not reach the allowable limit strain values of membrane only load and combined membrane and bending load.

3.8.1.4.13 Design Summary Report

A design summary report for the containment structures is presented in Appendix 3.8A where the design of representative critical sections of the structures is described.

The evaluation considering the deviations of as-procured or as-built construction to the design is performed with the acceptance criteria described in Subsection 3.8.1.5.

3.8.1.5 Structural Acceptance Criteria

The allowable stresses, strains, forces, displacements and temperatures for the containment structures including the liner are defined based on the requirements given in CC-3000 of the ASME Code, and NRC RG 1.136 and 1.216. When the containment structure is subjected to the load combinations described in Table 3.8-2, the allowable stresses, strains, forces or displacements specified below are not exceeded in order that:

a. The containment is essentially elastic under service load conditions.

b. General yielding of the reinforcing steel does not develop under factored primary load conditions.

c. The leak-tight integrity of the liner is maintained.
3.8.1.5.1 Acceptance Criteria for Service Load Conditions

The acceptance criteria under service load conditions are given below:

3.8.1.5.1.1 Concrete

Compression

The allowable stresses for the service load combinations defined in Table 3.8-2 are as follows:

a. Primary membrane stress: 0.30 $f'_c$

b. Primary membrane plus flexural stress: 0.45 $f'_c$

c. Primary plus secondary membrane stress: 0.45 $f'_c$

d. Primary plus secondary membrane plus flexural stress: 0.60 $f'_c$

Tension

Concrete tensile strength is not relied on to resist flexural or membrane tension.

Radial Shear

The allowable concrete stress and the limiting maximum stress for radial shear are in accordance with CC-3431.3 of the ASME Code.

Tangential Shear

The allowable concrete stress and the limiting maximum stress for tangential shear are in accordance with CC-3431.3 of the ASME Code.

Peripheral Shear

The allowable concrete stress and the limiting maximum stress for peripheral shear are in accordance with CC-3431.3 of the ASME Code.
Torsional Shear

The allowable concrete stress and the limiting maximum stress for torsional shear are in accordance CC-3431.3 of the ASME Code.

3.8.1.5.1.2 Prestressing System

Tendon

The tendon stresses during stressing and anchoring and the tendon stresses used for the design do not exceed the following:

a. During stressing, the tendon tensile stress at the anchor point does not exceed 0.80 $f_{pu}$ or 0.94 $f_{py}$, whichever is less.

b. Immediately after anchoring, the tensile stress at the anchor point does not exceed 0.73 $f_{pu}$, and the calculated average tensile stress over the length of the tendon does not exceed 0.70 $f_{pu}$.

c. For the purpose of design, the effective prestress is based on tendon stresses not exceeding those calculated to occur immediately after anchoring minus all applicable losses.

End Anchor

Compression under the tendon end anchor bearing plates is in accordance with the requirements of CC-3431.1 of the ASME Code.

Prestress Losses

To determine the effective prestress, the prestress losses are considered in accordance with CC-3542 of the ASME Code. The wobble friction coefficient and curvature friction coefficient are determined experimentally and verified during stressing operations. In addition, NRC RG 1.35.1 is used to establish the upper bound and lower bound of prestress losses at the time of 60-year design life.
3.8.1.5.1.3 Reinforcement Steel

**Tension**

a. The average tensile stress in reinforcing steel does not exceed 0.5 f_y.

b. The values given in a. above may be increased by 33.3 percent when the following loads are combined with other loads in the load combination:

1) Temporary loads from prestressing, which decrease after completion of prestressing (considered in construction load combination)

2) Secondary loads due to concrete volume change such as creep, shrinkage, and thermal growth

c. The values given in a. above may be increased by 50 percent when the temporary pressure loads during the test condition are combined with other loads in the load combination.

**Compression**

a. For load-resisting purposes, the compressive stress in reinforcing steel does not exceed 0.5 f_y.

b. The values given in a. above may be increased by 33.3 percent when the following loads are combined with other loads in the load combination.

1) Temporary loads from prestressing, which decrease after completion of prestressing (considered in construction load combination)

2) Temporary test pressure loads (considered in test condition)

3) Secondary loads due to concrete volume change such as creep, shrinkage, and thermal growth

**Radial Shear**

The radial shear reinforcement for service loads is provided in accordance with CC-3522 of the ASME Code.
Tangential Shear

The tangential shear reinforcement for service loads is provided in accordance with CC-3522 of the ASME Code.

Peripheral Shear

The peripheral shear reinforcement for service loads is provided in accordance with CC-3522 of the ASME Code.

Torsional Shear

The torsional shear reinforcement for service loads is provided in accordance with CC-3522 of the ASME Code.

Radial Tension Reinforcement

Radial tie reinforcement is provided to resist radial tensile forces from curved tendons in portions of the containment with double curvature in accordance with CC-3545 of the ASME Code.

End Anchor Reinforcement

Tendon end anchor reinforcement oriented perpendicular to the direction of applied force is provided to control cracking in the end anchor zone in accordance with CC-3543 of the ASME Code.

3.8.1.5.2 Acceptance Criteria for Factored Load Conditions

The acceptance criteria under factored load conditions are given below.

3.8.1.5.2.1 Concrete

The allowable stresses for the factored load combinations defined in Table 3.8-2 are as follows:

a. Primary membrane stress: 0.60 f_c

b. Primary membrane plus flexural stress: 0.75 f_c

c. Primary plus secondary membrane stress: 0.75 f_c
d. Primary plus secondary membrane plus flexural stress: 0.85 $f'_c$

**Tension**

Concrete tensile strength is not relied on to resist flexural or membrane tension.

**Radial Shear**

Radial shear is a transverse shear and is similar to shear in beam analysis. It occurs in the vicinity of discontinuities in shell flexural or membrane behavior. The allowable shear stress in prestressed concrete is determined in accordance with CC-3421.4 of the ASME Code.

**Tangential Shear**

Tangential shear is a membrane shear in the plane of the containment shell resulting from lateral load such as earthquake, wind, hurricane, or tornado loading. The allowable shear stress in prestressed concrete is determined in accordance with CC-3421.5 of the ASME Code.

**Peripheral Shear**

Peripheral shear stress is a transverse shear and is similar to punching shear in slab analysis. It is the shear resulting from a concentrated force or reaction acting transverse to the plane of the wall. The peripheral shear stress carried by concrete is in accordance with CC-3421.6 of the ASME Code.

**Torsional Shear**

Torsional shear stress is a local, in-plane shear stress induced in the containment wall due to torsional moment (from pipe penetrations or attachments) applied about an axis normal to the containment wall. The torsional shear stress carried by concrete is in accordance with CC-3421.7 of the ASME Code.

3.8.1.5.2.2 **Prestressing System**

**Tendons**

The maximum axial tensile capacity of the tendon is 90 percent of the yield strength in accordance with CC-3423 of the ASME Code.
Prestressing Losses

To determine the effective prestress, allowance for the following loss of prestress is considered.

a. Slip at anchorage
b. Elastic shortening of concrete
c. Creep of concrete
d. Shrinkage of concrete
e. Stress relaxation of tendon
f. Frictional loss due to intended or unintended curvature in the tendons

The loss of prestress due to friction losses is calculated in accordance with CC-3542 of the ASME Code. The wobble friction coefficient and curvature friction coefficient are determined experimentally and verified during stressing operations. In addition, NRC RG 1.35.1 is used to establish the upper bound and lower bound of prestress losses at the time of 60-year design life.

3.8.1.5.2.3 Reinforcing Steel

Tension

a. The design yield strength of reinforcing steel does not exceed 413.7 MPa (60 ksi).
b. The allowable stress for load resisting purpose does not exceed 0.9 $f_y$.
c. If more than one layer of reinforcement is provided to resist primary bending moment, the tensile strain in one or more of the layers may exceed 0.9 $\varepsilon_y$, provided a state of general yielding of the cross section is not reached.
d. The tensile strain in reinforcement around large openings may exceed 0.9 $\varepsilon_y$, provided the average strain for the total forces and moments over a distance of one-half the containment wall thickness from the opening or 25 percent of the opening diameter, whichever is smaller, does not exceed 0.9 $\varepsilon_y$. 
e. Under combined primary and secondary forces, the tensile strain in reinforcement may exceed $0.9 \varepsilon_y$. The maximum tensile strain does not exceed $2 \varepsilon_y$ in any reinforcement.

Compression

a. For load-resisting purposes, the compressive stress in reinforcing steel does not exceed $0.9 f_y$.

b. The compressive strains may exceed yield strength when acting in conjunction with concrete if the concrete requires larger strains than the reinforcing yield strain to develop its capacity.

Radial Shear

The radial shear reinforcement is provided in accordance with CC-3521.2 of the ASME Code.

Tangential Shear

The orthogonal tangential shear reinforcement is provided in accordance with CC-3521.1 of the ASME Code.

Peripheral Shear

The peripheral shear reinforcement is provided in accordance with CC-3521.3 of the ASME Code.

Torsional Shear

The torsional shear reinforcement is provided in accordance with CC-3521.4 of the ASME Code.

Radial Tension Reinforcement

Radial tie reinforcement is provided to resist radial tensile forces from curved tendons in portions of the containment with double curvature in accordance with CC-3545 of the ASME Code.
End Anchor Reinforcement

Tendon end anchor reinforcement oriented perpendicular to the direction of applied force is provided to control cracking in the end anchor zone in accordance with CC-3543 of the ASME Code.

3.8.1.5.3 Acceptance Criteria with Respect to Concrete Temperatures

The requirements pertaining to temperature limitations in concrete containments are in accordance with CC-3440 of the ASME Code.

a. For normal operation or any other long-term period, the temperatures are not to exceed 66 °C (150 °F) except for local areas, such as around a penetration, which are allowed to have increased temperatures not to exceed 93 °C (200 °F).

b. For accident or any other short-term period, the temperatures are not to exceed 177 °C (350 °F) for the interior surface. However, local areas are allowed to reach 343 °C (650 °F) from steam or water jets in the event of a pipe rupture.

3.8.1.5.4 Acceptance Criteria for Impactive and Impulsive Loading

Yield strain and displacement values are permitted to exceed general stress and strain limits due to impactive and impulsive loading. In the case of impulse loads, the usable ductility is 33 percent of the failure value and for impact effects, the usable ductility is 67 percent of the failure value in accordance with ASME Section III, CC-3920, for the design of the containment. Examples of impactive and impulsive loading include loading due to high-energy piping line breaks, localized yielding due to jet impingement and whip restraint loads, and external and internal missile loading. The design of containment internal structure is addressed in Subsection 3.8.3. General design for missiles is addressed in Section 3.5.

3.8.1.5.5 Acceptance Criteria for the Liner System

The acceptance criteria for the liner system are provided below.
Liner Plate

The acceptance criteria for the liner plate are the stress and strain limits specified in the ASME Section III, Table CC-3720-1, when considering the load combinations stated in Table CC-3230-1 with a load factor of 1.0.

Liner Anchors

The acceptance criteria for the liner anchors are the force and displacement allowable values given in ASME Section III, Table CC-3730-1.

Penetration Assemblies

The acceptance criteria are the design allowables given in ASME Section III, CC-3740 and CC-3820. In accordance with CC-3740 (b), the design allowables for penetration nozzles are the same as used for ASME Section III, Division 1, where a nozzle is defined as that part of the penetration assembly not backed by concrete.

In accordance with Subarticle CC-3740 (c), the design allowables for the liner in the vicinity of the penetration are the same as those given in AISC N690 for resisting mechanical loads in the service load category. For factored load categories, the allowables are increased by a factor of 1.5, except for impulse loads and impact effects.

In accordance with Subarticle CC-3740(d), the portion of the penetration sleeves backed by concrete is designed to meet the acceptance criteria described above for the liner plate and anchors. Additionally, consistent with requirements in Subarticle CC-3820, to verify acceptability, the structural capacities of penetration assemblies that are designed for pipe loads are compared against (a) the ultimate moment, axial, torque, and shear loadings that the piping is capable of producing or (b) penetration loads based on a dynamic analysis considering pipe rupture thrust as a function of time. In (b), penetration designs are later verified using results of piping analysis to provide reasonable assurance that the load used in the design is not exceeded.

Typically for the APR1400, in order to preclude pipe rupture effects, flued heads are used for high-energy piping if large pipe rupture design loads are anticipated. See Section 3.6 for further details on this topic.
Brackets and Attachments

The allowables given in the ASME Section III, CC-3650 and CC-3750, are used as the acceptance criteria for brackets and attachments to the liner.

The APR1400 design avoids the use of brackets and similar items that transmit loads to the liner in the through-thickness direction. As much as practical in the design of attachments that have structural components carrying major loads, for example the upper plates of crane brackets, such a structural component of the attachment is made continuous through the liner. When through-thickness liner loads cannot be avoided and the liner is 25 mm (1 in.) or more thick, then the special welding and material requirements of Subarticle CC-4543.6 are applied. In addition to the requirements given in Subarticle CC-4543.6 (a) through (d), ultrasonic examinations are required prior to fabrication to preclude the existence of laminations in the installed material.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

This section contains information relating to the materials, quality control program, and special construction techniques and tolerances used in the fabrication and construction of the containment. Materials and quality control satisfy the following requirements:

a. ASME 2001 Edition with 2003 Addenda, Section III, Division 2, Code for Concrete Containments, CC-2000, CC-4000, CC-5000, and CC-6000


Concrete and reinforcement forming and placement tolerance not specifically addressed in these references are in accordance with ACI 349 and ACI 117. Any provisions in codes which provide acceptance criteria for conditions when to tolerances are exceeded shall not apply, but a licensee referencing the APR1400 DCD may deviate from the tolerances in these codes using alternative acceptance criteria, if these acceptance criteria are approved by the NRC. As-built dimensions will also be evaluated to verify compliance with the design bases and applicable codes and standards identified in Table 3.8-1. The COL applicant may provide construction tolerance acceptance criteria and the basis for the criteria (e.g., through the use of analysis, industry research, or testing) for cases where the tolerances in the ACI 117 and ANSI/AISC 303, for structural concrete and structural steel, respectively, may be exceeded (COL 3.8(22)).
3.8.1.6.1 Concrete and Concrete Ingredients

The materials to be used for concrete and concrete ingredients are given below.

Cement

Cement used for the concrete containment conforms with the requirements of ASTM C150 (Reference 10).

Aggregates

Aggregates used for the concrete containment conforms with the requirements of ASTM C33, (Reference 11) with the requirements specified in CC-2222 of the ASME Section III, Division 2.

Mixing Water

Mixing water used for the concrete containment conforms with the requirements of CC-2223 of the ASME Section III, Division 2.

Admixtures

Air-entraining admixtures used for the concrete containment conform with the requirements of ASTM C260 (Reference 12). Chemical admixtures used for the concrete containment conform with the requirements of ASTM C494 (Reference 13). Mineral admixtures (fly ash and pozzolan) used for the concrete containment conform with the requirements of ASTM C618 (Reference 14).

Concrete Mix Design

The concrete mix design for the concrete containment conforms with the requirements of CC-2230 of the ASME Section III, Division 2.

Concrete Compressive Strength

The specified minimum compressive strength of 41.37 MPa (6,000 psi) at 91 days is used for the containment wall and dome. For the containment common basemat, the specified minimum compressive strength of 34.47 MPa (5,000 psi) at 91 days is used.
3.8.1.6.2 Reinforcing Bars and Splices

The material to be used for reinforcing bars conforms with ASTM A615 (Reference 15) and the requirements described in CC-2330 of ASME Section III, Division 2.

The mechanical splices conform with the permitted types described in CC-4331.2 of ASME Section III, Division 2. The material to be used for bar-to-bar splice sleeves in reinforcing bars conforms with ASTM A513, A519, or A576 (References 16, 17, and 18, respectively). The material to be used for reinforcing bar splice sleeves attached to liner plates or structural steel shapes is a carbon steel conforming to ASTM A513, A519, or A576, Grades 1008 through 1030.

3.8.1.6.3 Prestressing System

The prestressing system of the concrete containment is VSL E6-42, multi-strand system using wedge block with wedge type anchors. The material to be used for prestressing system is given below.

Prestressing Steel

The material for prestressing elements conforms with ASTM A416 (Reference 19) and the requirements described in CC-2420 of ASME Section III, Division 2.

Anchorage Components

Materials for anchorage components such as bearing plates, anchor head assemblies, and wedges conform with the tendon manufacturer’s respective material specifications. In addition, the materials for anchorage components conform with the requirements described in CC-2430 of ASME Section III, Division 2 and NRC RG 1.136.

Non-Load-Carrying and Accessory Materials

Non-load-carrying materials such as tendon duct, channel, trumpet, and transition cones shall be ferrous in conformance with the requirements in CC-2441 of ASME Section III, Division 2. The temporary and permanent corrosion prevention materials conform with the requirements of CC-2442 of ASME Section III, Division 2.
3.8.1.6.4 Liner Plate within Containment Backed by Concrete

The materials, fabrication procedures, and examination requirements conform with the technical provisions of CC-2500, CC-4500, and CC-5500 of ASME Section III, Division 2.

The containment liner materials backed by concrete meet the requirements of CC-2500 of ASME Section III, Division 2, and conform with the following specifications:

<table>
<thead>
<tr>
<th>Application</th>
<th>Specification</th>
</tr>
</thead>
<tbody>
<tr>
<td>Liner plate, embedment plate</td>
<td>SA-516 Grade 55, 60, or 70</td>
</tr>
<tr>
<td></td>
<td>SA-240</td>
</tr>
<tr>
<td>Liner anchor</td>
<td>SA-36</td>
</tr>
</tbody>
</table>

The material to be used for containment wall liner at El. 78 ft through El. 101 ft conforms with ASME SA240 (Reference 20) and the requirements described in CC-2500 of the ASME Section III, Division 2.

The fabrication of the containment steel boundaries backed by concrete is in accordance with CC-4500 of ASME Section III, Division 2. The qualifications of welders and welding procedures are in accordance with CC-4530 of ASME Section III, Division 2.

All nondestructive examination procedures are in accordance with Section V of the ASME Code.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Integrity Test

The structural integrity test (SIT) is performed in accordance with ASME Section III, Division 2, CC-6000 to verify the structural integrity of the containment. The test is performed after the containment is complete, including the liner, concrete structures, all electrical and piping penetrations, equipment hatch, personnel airlocks, and post-tensioning.

The pressure will be brought up to 115 percent of the containment design pressure in approximately five or more equal increments. At each pressure level, the pressure will be held constant for 1 hour prior to measuring the deflections.
3.8.1.7.1.1 General Requirements

Prior to operating the plant, the SIT is performed to demonstrate the structural acceptability of the primary containment.

At each pressure level, the pressure will be held constant for 1 hour to allow for the pressure and containment response to stabilize and to survey the exterior surface of the primary containment. Internal and external temperature measurements will be taken.

Under each test pressure level, the crack patterns are measured in accordance with CC-6225 of ASME Section III, Division 2. All cracks larger than 0.254 mm (0.01 in.) and longer than 150 mm (6 in.) will be mapped at the locations required in CC-6350 of ASME Section III, Division 2.

Displacement of the containment is measured and evaluated in conformance with CC-6223 and CC-6360 of ASME Section III, Division 2.

Pre- and post-test inspections will be performed to confirm that the concrete, liner plate, and interior structures were not damaged during the SIT.

3.8.1.7.1.2 Response Prediction Prior to the Structural Integrity Test

The expected readings of all devices used to monitor containment behavior are determined before the SIT.

The predictions of the containment responses under test pressure will be obtained using the same analytical models, computer programs, and analysis procedures that are used in the containment design so the adequacy of all analytical results is confirmed by the test.

The liner plate is not permitted to be used as a strength element in designing the containment to resist accident pressure or earthquake loads. However, during the SIT, the liner plate is in tension and is considered in the prediction of response.

Acceptance limits will be developed based on the analytical predictions for each pressurization increment to permit a judgment as to whether the test may proceed safely to the next increment of increased pressure.
3.8.1.7.1.3 Acceptance Criteria

a. Overall structure exhibits elastic behavior.

b. No visible signs of permanent damage to either the concrete structure or the steel liner are detected.

c. Residual displacements at the points of maximum predicted radial and vertical displacement at the completion of depressurization or up to 24 hours later do not exceed 20 percent of measured or predicted displacement at maximum test pressure, whichever is greater, plus 0.25 mm (0.01 in.) plus measurement tolerance. The above criteria are applied to the average of radial displacements measured at the same elevation.

d. When the measured displacements at the points of maximum predicted radial and vertical displacement do not exceed the measurement tolerance plus 30 percent of the measured or predicted displacement at the maximum test pressure, the above criteria are applied to the average of radial displacements measured at the same elevation. This criterion may be waived if the residual displacements within 24 hours are not greater than 10 percent.

Test results and conclusions will be documented in a separate report.

3.8.1.7.2 Inservice Surveillance

3.8.1.7.2.1 General Requirements

During the plant life, the in-service inspection of the containment is performed in accordance with the requirements of the ASME Section XI, IWL (Reference 4), NRC RG 1.35.1, and 10 CFR 50.55a. The in-service inspection includes a visual examination of the concrete exterior surface for cracking, spalling, or grease leakage; a visual inspection of the tendon anchorage assembly of sampled tendons; a tendon lift-off test to discover damaged or broken tendon wires and to provide reasonable assurance of an acceptable prestress level during the plant life.

The visual examination is performed for accessible areas in accordance with IWL-2510 of ASME, Section XI. When conditions exist in accessible areas that could indicated the presence of, or result in, degradation to inaccessible areas, the acceptability of inaccessible areas will be evaluated to meet the requirements of 10 CFR 50.55a.
The examination of the post-tensioning system, including sample selection, tendon force and elongation measurements, and examination of anchorage areas meets the requirements of IWL-2520 of ASME, Section XI.

The containment is designed to allow access to the post-tensioning systems during in-service inspections.

The containment leakage tests comply with 10 CFR Part 50, Appendix J and are performed in accordance with NEI 94-01 and ANSI/ANS 56.8. Type A, B, and C tests are described in Section 6.2.6.

3.8.1.7.2.2 Prediction of Minimum Lift-off Forces

The prediction of the minimum tendon lift-off forces is made prior to in-service inspection.

The prediction of tendon lift-off forces is determined based on the estimated prestress losses at the time of test. As a minimum, the following sources of prestress loss are considered:

a. Elastic shortening taking into consideration the sequence of stressing of the tendon

b. Creep and shrinkage of concrete

c. Stress relaxation in tendon

d. Reduction of wire cross section due to corrosion, if any

e. Ineffective wires (e.g., broken, unseated) documented during initial stressing

The procedures and formulations presented in NRC RG 1.35.1 are used to establish the upper bound and lower bound of prestress losses at the time of test.

3.8.1.7.2.3 Acceptance Criteria

The acceptance criteria for the in-service inspections are as specified in ASME Section XI, IWL-3000.

Items with examination results that do not meet the acceptance standards are evaluated to determine:
3.8.2 Steel Containment

The APR1400 does not use a steel containment.

This subsection pertains to ASME Class MC components that are part of the containment described in Subsection 3.8.1. ASME Class MC components include the equipment hatch, personnel airlocks, piping and electrical penetrations, and fuel transfer tube sleeve and bellows.

3.8.2.1 Description of Containment

3.8.2.1.1 Equipment Hatch

The equipment hatch consists of a dished head door, matching barrel frame with anchorage, and the lifting equipment that operate the hatch door as shown in Figure 3.8-13. The clear opening of the equipment hatch has an inside diameter of 7.92 m (26 ft). The hatch is located at centerline elevation of 167 ft 6 in.

The equipment hatch is provided for access to the interior of the reactor containment building during shutdown. The transfer of equipment through the containment wall is accomplished through this hatch. The equipment hatch will be a round barrel frame with a dished head access hatch. The dished head can be fully removed with a lifting bracket located near the center of gravity of the entire removable assembly.

3.8.2.1.2 Personnel Airlocks

A typical personnel airlock is shown in Figure 3.8-14. Two personnel airlocks are provided. Each personnel airlock has an inside diameter of 3.05 m (10 ft) to accommodate a door that is 1.07 m (3 ft 6 in.) wide and 2.03 m (6 ft 8 in.) high.

Personnel airlocks consist of doors with two gaskets in series, which are mechanically interlocked so that one door cannot be opened unless the second door is closed and sealed.
If needed, the mechanical interlock can be overridden by use of a special procedure provided. The doors operate manually. Either door is operated from inside the reactor containment building, inside the access hatch, or outside the reactor containment building. Each door is equipped with a valve for equalizing pressure on both sides of the door before the door can be operated.

Valves are operated at the same location where the associated door can be operated. The valves are interlocked so that only one valve can be operated at a time and only when the opposite door is closed and sealed. Indicators are located on the outside of the access hatch at each door to show whether the opposite door and its valves open or close. The access hatch pressure can be tested any time without interference in the normal operation of the plant. Provisions are made for continuous leak testing of the door seals on both doors.

3.8.2.1.3 Penetrations

Penetrations are provided in the containment wall to:

a. Extend process piping and electrical conductors through the containment wall
b. Provide reasonable assurance of the integrity of the containment
c. Act as supports for the process pipelines

3.8.2.1.3.1 Penetration Types

Penetrations consist of two major types:

a. Process pipe penetrations
b. Electrical penetrations

3.8.2.1.3.1.1 Process Pipe Penetrations

Process pipe penetrations include instrumentation, HVAC, and mechanical piping penetrations. A typical penetration assembly consists of some or all of the following components:

a. Sleeve embedded in concrete wall
b. Sleeve anchors
c. Head fitting

d. Penetration seals (air, water, and radiation)

The process pipe penetrations used are the three types shown in Figure 3.8-8.

Determination of penetration types is made during design process based upon the magnitude of applicable loads and the in-service inspection requirements.

3.8.2.1.3.1.2 Electrical Penetrations

Electrical penetration assemblies are used to extend electrical conductors through the pressure boundary of the reactor containment building. Electrical penetrations are functionally grouped into medium-voltage power, low-voltage power, low-voltage control, and instrument cable penetration assemblies. Figure 3.8-9 shows a typical electric penetration assembly in place within the containment wall. An assembly is sized to be inserted into sleeves in the containment wall.

3.8.2.1.3.2 Component Classification

The portions of the penetration sleeves not backed by concrete are designed as a class MC component in accordance with ASME Section III, Division 1, NE. The portions of the penetration sleeves backed by concrete are designed to meet the requirement of ASME Section III, CC-3700 and CC-3800.

All penetration head fittings (penetration Type 1 of Figure 3.8-8) are classified as piping components and, as such, they have the same classification as the process pipe and are designed in accordance with ASME Section III, Division 1, NB, NC, or ND as applicable. The other head fittings are designed as ASME Class MC components.

3.8.2.1.4 Fuel Transfer Tube Sleeve and Bellows

Nuclear fuel is supplied into the reactor containment building and spent fuel is taken out through the fuel transfer tube penetration. The fuel transfer tube sleeve connects the refueling pool in the reactor containment building and the spent fuel pool in the auxiliary building. The sleeve wraps around the fuel transfer tube and the bellows are installed in case of differential movement of the two buildings. The bellows are welded to the penetration sleeve after installation of the transfer tube. The sleeve and bellows are made
of stainless steel. Conceptual fuel transfer tube sleeve and bellows are shown in Figure 3.8-25.

3.8.2.2 Applicable Codes, Standards, and Specifications

The following regulations, codes, standards, and specifications are used in the design of the class MC components.

3.8.2.2.1 Design Codes, Standards, Specifications, and Regulations

The design codes, standards, specifications, and regulations are listed in Table 3.8-1. The primary design code for class MC components is ASME Section III, Division 1, NE (Reference 2).

3.8.2.2.2 NRC Regulatory Guides

Conformance to each NRC Regulatory Guide is described in Section 1.9. NRC RG 1.7 (Reference 7) and NRC RG 1.57 (Reference 21) are applicable to the design of the ASME Class MC components.

3.8.2.2.3 Industry Standards

Nationally recognized industry standards, such as those published by ASTM, are used whenever possible to describe material properties, testing procedures, and fabrication and construction methods.

3.8.2.3 Loads and Load Combinations

The loads and load combinations for class MC components are given in Table 3.8-3. These loads and their combinations conform with ASME Section III, Division 1, NE 3000.

The terms of design loads are defined in Subsection 3.8.1.3.

3.8.2.3.1 Loads for Instrument and Process Pipe Penetrations

The forces and moments imposed at the piping penetration assembly boundaries are due to the following:

a. Internal and external operating and design pressures and temperatures
b. Process pipe reactions due to (as applicable):

1) Weight

2) Safe shutdown earthquake (SSE)

3) Thermal expansion

4) Relative dynamic displacements

5) Fluid dynamics (e.g., pressure, temperature, and hydraulic transients; main steam safety valves)

6) Pipe rupture and jet impingement

c. Severe accident load due to (as applicable):

1) Pressure load generated from 100 percent fuel clad metal-water reaction

2) Pressure loads generated by hydrogen burning

3.8.2.3.2 Load Combinations for Instrument and Process Piping Penetrations

The instrument and process piping penetration assemblies are composed of pipe components and/or MC components. The load combinations for each component of the penetration assemblies are defined to conform to the Code requirements relevant to the applicable Code for each component. Load combinations for ASME Class MC components, such as sleeves and head fittings (penetration Type 2 & 3 of Figure 3.8-8) are given in Table 3.8-3. Load combinations for penetration head fittings of Type 1 from Figure 3.8-8 are given in Table 3.12-1 or Table 3.12-2.

3.8.2.4 Design and Analysis Procedures

3.8.2.4.1 Equipment Hatch, Personnel Airlocks, and Electrical Penetrations

The equipment hatch, personnel airlocks, and electrical penetrations are designed as pressure-retaining components. The portions of the sleeves not backed by concrete are analyzed and designed according to the provisions of ASME Section III, Division 1, NE 3000.
The equipment hatch, personnel airlocks, and electrical penetrations are vendor designed components. The COL applicant is to provide detailed analysis and design procedure for the equipment hatch, personnel airlocks, and electrical penetrations (COL 3.8(3)). The major design aspects and criteria for the equipment hatch and personnel airlocks are as follows.

The equipment hatch design consists of a flanged cover bolted to a matching flanged cylindrical sleeve embedded into the reactor containment building. The equipment hatch closure head and sleeve shall be evaluated for design and service conditions in accordance with the requirements of NE 3000. The closure head spherical shell thickness shall be adequate for the design pressure in accordance with the rules of NE 3133.4. The head flange and sleeve flange shall meet the rules of NE 3326.2 and Appendix XI 3260. The swing bolts shall provide adequate bolt area for all preload, design and service conditions.

The personnel airlock design consists of a cylindrical shell having a bulkhead and pressure retaining door at each end. The personnel airlock pressure retaining components shall be evaluated for design and service conditions in accordance with the requirements of NE 3000. The airlock door plate thickness and stiffeners, and the bulkhead plate and stiffeners shall be adequate for the design condition in accordance with the stress limits of NE 3221. The airlock shell thickness shall meet the requirements of NE 3133 and NE 3324 for the external and internal pressure conditions.

3.8.2.4.2 Process Piping Penetrations

The entire penetration assembly including the sleeve, head fitting, and attached portion of pipe is designed for the loads described in Subsections 3.8.2.3.1 and 3.8.2.3.2 by the finite element computer program ANSYS. The penetration assemblies provide for process piping to pass through the containment structure or other building walls or floors.

Each penetration assembly may consist of the following components.

a. Portion of the process or instrument piping.

b. The penetration sleeve, which encloses the process piping and which is embedded and anchored in the wall or floor.
c. The head fitting (forged flued head or flat plate ring), which is welded to both the process pipe and the penetration sleeve. A typical piping penetration assembly configuration is shown in Figure 3.8-8.

Finite element analysis programs for the axisymmetric structural analysis for the stress analysis of the piping penetration assembly are used. The model will generate an automatic mesh, and imposed load and thermal condition are input to the model to perform FEM structural analysis. The pipe outside diameter, pipe wall thickness, sleeve outer diameter, sleeve wall thickness, head fitting thickness, and insulation thickness are considered in the analysis model.

Boundary conditions can be classified into three categories for the performance of the stress analysis of the piping penetration assembly; 1) stress boundary at the penetration assembly section connected to pipe where the stress distribution is applied corresponding to its location against imposed force, 2) inner surface of the pipe and sleeve where pressure is acting normal to the surface, and therefore stresses normal to the surfaces are applied as pressure for the boundary condition, and 3) fixed point at the concrete wall.

The forces and moments imposed at the penetration assembly boundaries are due to the following:

a. Internal and external operating and design pressures and temperature

b. Process pipe reactions due to (as applicable):
   1) Weight
   2) Safe shutdown earthquake (SSE)
   3) Thermal expansion and relative dynamic displacements
   4) Hydraulic transients and other non-seismic dynamic loads
   5) Pipe rupture and jet impingement

c. Maximum reactor containment building pressure during severe accidents
The penetration assemblies are required to meet the stress limits under the worst loading combinations for testing, design, and Service Conditions for Level A, B, C, and D service limit in accordance with the requirements and provisions of ASME Code Section III.

Loading combinations are formulated in a way that will assure the highest resultant stress. For the sleeve anchors, the types of loads applicable for each condition will be as stated in Table CC-3230-1 of ASME Section III, Division 2. For the ASME Class MC components, the applicable loads for each condition are as listed below and Table 3.8-3

a. Testing Condition: Test pressure and temperature plus dead load and live load

b. Design Condition: These include all design loadings for which the containment vessel or portions thereof might be designed during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design-basis LOCA (as applicable).

c. Service Condition for Level A and B service limits: These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design basis accident conditions for which the containment function is required. The additional loads resulting from natural phenomena during which the plant must remain operational for Level B service limits.

1) For Expansion Stress Evaluation: Loads due to operating temperature, thermal expansion and relative dynamic displacements

2) For Primary-plus-Secondary Stress Evaluation: Pressure, temperature, and reaction load, plus mechanical loads due to: dead load, live load, thermal expansion and relative dynamic displacements, hydraulic transients such as actuation of SRV discharge and subsequent hydrodynamic reaction loads, OBE, and LOCA, as applicable

d. Service Condition for Level C service limits: These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required.

Pressure, temperature, and reaction load, plus loads due to: dead load, live load, thermal expansion, hydraulic transients such as actuation of SRV discharge and
subsequent hydrodynamic reaction loads, SSE, and LOCA, pressure resulting from hydrogen release are applied as applicable.

e. Service Condition for Level D service limits: These service limits include other applicable service limits and loadings of a local dynamic nature for which the containment function is required.

1) Pressure, temperature, and reaction load, plus loads due to: dead load, live load, thermal expansion, hydraulic transients such as actuation of SRV discharge and subsequent hydrodynamic reaction loads, SSE, and LOCA, pipe rupture and jet impingement are applied as applicable.

2) Maximum operating pressure applied in the annulus between the process pipe and the penetration sleeve for process piping penetration assembly

f. Post-flooding Condition: This includes the post-LOCA flooding of the containment in combination with OBE-basis earthquake

g Fatigue evaluation: For the metal fatigue evaluation, pressure- and load-range combinations must be formed in a way that will assure cyclic stress optimization. For the formation of these load-range combinations the loads listed in Subsection above c) 2 will be used. Contribution from thermal transients will also be considered. With the exception of the pressure and temperature values, the numerical values of the load components for each penetration assembly are obtained from appropriate stress analysis reports for the corresponding piping systems. Fatigue evaluation for MC component is performed in accordance with NE-3221.5 of the ASME Section III, Division 1.

The mechanical forces and moments are applied to the penetration assembly at each boundary. In addition to the mechanical loads, loads due to the design and operating pressure within the pipe penetration annulus are applied. The loads and their combination methods are specified in the project unique penetration specification. These loads consist of the design and operating pressure, weight, earthquake, hydraulic transients, SRV, condensation oscillation, chugging, thermal expansion, and relative seismic displacements. Stress optimization is achieved by analyzing the penetration for the worst load combination for each loading condition (i.e., Design, Level A and B service limit, and Level C and D service limit).
It is required that the penetration assemblies be analyzed for the various design and operating component conditions, and that the resulting stresses be within the specified allowable limits. The allowable stress limits associated with each service load category are as stated below and Table 3.8-3.

a. Stress Limits for Process Piping: In accordance with the Piping System Design Specification

b. Stress Limits for Penetration Sleeve Anchors: In accordance with Division 2 of the ASME Section III.

c. Stress Limits for Penetration Sleeves and Head Fittings: For MC penetration sleeves and head fittings the allowable stress limits shall be as specified in NE-3200 of the ASME Section III, Division 1, for Class MC components.

3.8.2.4.3 Fuel Transfer Tube Sleeve and Bellows

The fuel transfer tube penetration assembly is supplied by a vendor. The COL applicant is to provide detailed analysis and design procedure for the transfer tube penetration assembly (COL 3.8(4)). The fuel transfer tube sleeve and bellows shall be designed as Class MC components in accordance with ASME Section III, Division 1, NE.

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 Equipment Hatch, Personnel Airlocks, and Electrical Penetrations

The equipment hatch, personnel airlocks, and electrical penetrations are designed as Class MC components according to ASME Section III, Division 1, NE.

Stress intensities for MC components other than bolts are limited to the values defined by ASME Section III, Division 1, NE-3221. Buckling stresses are limited to the values defined by ASME Section III, Division 1, NE-3222, while primary, secondary, and peak stresses are defined by ASME Section III, Division 1, NE-3213. Stress limits for bolts and allowable stress for test is in accordance with ASME Section III, Division 1, NE-3230 and NE-3226, respectively. The thickness of components under external loading is determined in accordance with the rules defined by ASME Section III, Division 1, NE-3133. Load combinations and their corresponding service levels to determine the appropriate stress limits are given in Table 3.8-3.
3.8.2.5.2 Process Piping Penetration Assemblies

The process piping penetrations are designed in accordance with ASME Section III, Division 1, NE.

Stress intensities are limited to the values defined by ASME Section III, Division 1, NE-3220. Load combinations and their corresponding service levels to determine the appropriate stress limits are given in Table 3.8-3.

3.8.2.5.3 Fuel Transfer Tube Sleeve and Bellows

The fuel transfer tube penetration sleeve and bellows are designed as class MC components in accordance with ASME Section III, Division 1, NE.

Stress intensities for the penetration sleeve are limited to the values defined by ASME Section III, Division 1, NE-3211. The fuel transfer tube sleeve shall be evaluated for the worst combination of design load, as applicable. Under the loading condition, the sleeve shall meet all applicable stress requirements set forth in ASME Section III, Division 1, NE-3221. The thickness of the sleeve shall be evaluated for design pressure in accordance with ASME Section III, Division 1, NE-3324.3. For the sleeve bellows, the requirements of NE 3366 shall be met for the expansion joint assembly.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The materials conform with the requirements of ASME Section III, Division 1, NE-2000 and physical properties for the materials are shown in Table 3.8-4.

The testing of the containment building leak-tight boundaries not backed by concrete is in accordance with ASME Section III, Division 1, NE.

All welds between the penetration and head fitting and between the penetration reinforcing plate and frames for airlocks and access openings are examined radiographically.

All welds in bellows-type expansion joints in penetration assemblies or appurtenances to the containment vessel are examined either by magnetic particle examination for ferrite material or by liquid penetrant examination for austenitic material.

The qualifications of welders and welding procedures are in accordance with ASME Section III, Division 1, NE-4300.
The physical properties of these materials are listed in Table 3.8-4. The penetration components mentioned in Subsection 3.8.1.6 fully conform with the materials specified in ASME Section III, Division 1, NE-2000.

The fabrication and installation requirements of ASME Section III, Division 1, NE-4000, and the provisions of ASME Section III, Division 1, NE-5000 and NE-6000, dealing with examination and testing of components are to be met. Standard construction techniques are used in the fabrication and erection of ASME Class MC components.

3.8.2.7 Testing and Inservice Inspection Requirements

3.8.2.7.1 Structural Acceptance and Initial Leak Rate Tests

All MC components are tested for their structural acceptance and leak rate at the same time of the containment tests described in Subsection 3.8.1.7.

Leak rate tests are performed on all hatches by pressurizing the plenum between the double gaskets. In addition, the personnel airlock will be shop tested according to the following procedure:

a. Pressure tests are performed for hatches in accordance with criteria specified in ASME Section III, Division 1, NE-6000.

b. Initial leak rate test: hatches are tested at a peak pressure. Leakage does not exceed the design limit of the volume of the airlock in 24 hours.

3.8.2.7.2 Inservice Inspection

MC components are tested in accordance with ASME Section III, Division 1, NE-6000.

Inservice inspections of the MC components follow the requirements of ASME Section XI, IWE, with the additional requirements of 10 CFR 50.55a. The visual examination is performed for accessible areas in accordance with IWE-2310 of ASME, Section XI. When conditions exist in accessible areas that could indicate the presence of or the result in degradation to inaccessible areas, the acceptability of inaccessible areas will be evaluated to meet the requirements of 10 CFR 50.55a. The inservice inspection summary report as required by IWA-6000 is to include the evaluation results for inaccessible areas identified for evaluation. The containment leakage tests comply with 10 CFR Part 50, Appendix J.
and are performed in accordance with NEI 94-01 and ANSI/ANS 56.8. Subsection 6.2.6 describes leak-rate testing of the containment system and associated acceptance criteria.

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

3.8.3.1 Description of the Internal Structures

The internal structure is a group of reinforced concrete structures that enclose the reactor vessel and primary system. The internal structure provides biological shielding for the containment interior. A description of various structures that constitute the internal structure is given in the following paragraphs. The details of the internal structure are shown in Figures 1.2-2 through 1.2-8.

The internal structures are seismic Category I structures with the exception of platforms that do not support seismic Category I equipment and miscellaneous steel.

Platforms that do not support seismic Category I equipment and miscellaneous steel are seismic Category II structures. Seismic Category II structures are designed for the SSE using seismic Category I criteria to prevent adverse interaction with other seismic Category I structures, systems, and components.

The containment internal structures are connected with the containment basemat by mechanical splices. The reinforcing bar of the internal structures is anchored into the foundation by the use of mechanical splices welded to both sides of a thickened liner plate. The layout of the connections between the internal structures and the foundation is shown in Figures 3.8A-16 and 3.8A-17. And Figure 3.8-7 (3 of 3) provides a typical connection detail.

The internal structures located in the reactor containment building consist of the following major components:

a. Reactor support system

b. Steam generator support system

c. Reactor coolant pump support system

d. Pressurizer support system
e. Primary shield wall (PSW) and reactor cavity

f. Secondary shield wall (SSW)

g. Refueling pool

h. In-containment refueling water storage tank (IRWST)
i. Holdup volume tank

j. Operating and intermediate floors

k. Interior concrete fill slab

l. Polar crane supports

3.8.3.1.1 Reactor Support System

The reactor vessel is supported by four columns under the cold leg nozzles (elevation 35.76 m (117 ft 4 in.)), as described in Subsection 5.4.15.2, which interface with anchor bolts embedded in the primary shield wall. Lateral supports are provided for the reactor vessel to resist horizontal loads. The lateral supports transmit the loads to the reactor cavity wall, which houses the reactor. In addition, shear keys at the lower part of the reactor vessel fit into the keyways located in the base plate (elevation 28.74 m (94 ft 3.5 in.)) of the column supports. The keyways, which are described in Subsection 5.4.15.2, transmit the horizontal loads to the cavity wall through shear bars attached to the bottom of the base plate. Both the lateral supports and shear keys are designed to allow movement that is the result of thermal growth of the reactor vessel in the radial and vertical directions. Reactor vessel supports are shown in Figure 3.8-15. Subsection 5.4.15 provides further information on the design of the reactor vessel support structure.

3.8.3.1.2 Steam Generator Supports

The steam generator is supported at the bottom by a sliding base bolted to an integrally attached conical skirt at elevation 34.53 m (113 ft 3.34 in.). The sliding base rests on low-friction spherical head bearings at elevation 34.21 m (112 ft 2.68 in.), which allows unrestrained thermal expansion of the reactor coolant system. Two keyways in the sliding base mate with keys in a forged plate installed on the concrete pedestal by anchor bolts to guide the movement of the steam generator during expansion and contraction of the reactor.
coolant system and to limit movement of the bottom of the steam generator during seismic, IRWST discharge, and BLPB events. Keys and snubbers on the steam drum guide the top of the steam generator at centerline elevations 51.672 m (169 ft 6.35 in.) and 48.399 m (158 ft 9.48 in.), respectively, during expansion and contraction of the reactor coolant system and provide support during seismic, IRWST discharge, and BLPB events. Typical steam generator supports are shown in Figure 3.8-16. Subsection 5.4.15 provides further information on the design of the steam generator support structures.

3.8.3.1.3 Reactor Coolant Pump Support System

Reactor coolant pump supports consist of four vertical columns at elevation 31.33 m (102 ft 9.5 in.), two horizontal snubbers, two upper horizontal columns at elevation 40.05 m (131 ft 4.75 in.), and two lower horizontal columns at elevation 34.82 m (114 ft 2.75 in.). These rigid structural columns provide support for the pumps during normal operation, earthquake conditions, IRWST discharge, and BLPB. Each column, horizontal and vertical, and the snubber end in a drilled clevis to accept anchor bolts. Support loads are transmitted to the concrete structures by anchor bolts. Typical reactor coolant pump supports are shown in Figure 3.8-17. Subsection 5.4.15 provides further information on the design of the reactor coolant pump support structures.

3.8.3.1.4 Pressurizer Support System

The pressurizer is supported by a cylindrical skirt as shown in Figure 3.8-18. This skirt is welded to the pressurizer and is anchored to the concrete slab at elevation 41.71 m (136 ft 10.25 in.). The skirt is designed to withstand deadweight and normal operating loads as well as the loads due to earthquakes, pressurizer POSRV actuation, IRWST discharge, and BLPB events. Support loads are transmitted to the concrete slab through the skirt flange and anchor bolts. Four keys welded to the upper shell of the pressurizer at elevation 54.52 m (178 ft 10.38 in.) provide an additional restraint for earthquake, pressurizer POSRV actuation, and BLPB conditions. Four keys in the pressurizer mate with keyways that are installed in the secondary shield wall by anchor bolts as shown in Figure 3.8-18. Subsection 5.4.15 provides further information on the design of the pressurizer support structures.

3.8.3.1.5 Primary Shield Wall and Reactor Cavity

The primary shield wall is a heavily reinforced concrete structure that houses the reactor vessel, provides the primary radiation shielding, and provides protection for the reactor vessel and its components.
vessel from internal missiles. It is anchored to the containment basemat through the use of mechanical splices welded to both sides of the thickened liner plate. The massive primary shield walls provide a support for the refueling canal walls above the reactor cavity. In plan, the primary shield walls form a monolithic ring, housing the reactor vessel. Penetrations in the primary shield wall are provided for the primary loop. The primary shield wall is designed to withstand the temperatures and pressures following a LOCA. The primary shield wall is a minimum of 2.01 m (6 ft 7 in.) thick.

3.8.3.1.6 Secondary Shield Wall

The secondary shield wall is a reinforced concrete structure surrounding steam generators, reactor coolant pumps, and pressurizer. The secondary shield wall protects the reactor containment building from internal missiles. In addition to providing a biological shield for the coolant loop and equipment, the secondary shield wall also provides structural support for pipe supports/restraints and platforms at various levels. The secondary shield wall is a right cylinder with an inside diameter of 29.87 m (98 ft) and a height of 34.44 m (113 ft) from its base. The secondary shield wall is a minimum of 1.22 m (4 ft) thick.

The secondary shield wall is anchored into the basemat by mechanical splices welded to both sides of the thickened liner plate.

3.8.3.1.7 Refueling Pool

The refueling pool filled with borated water facilitates the fuel handling operation without exceeding the acceptable level of radiation inside the reactor containment building. The inside of refueling pool is lined with a stainless steel liner plate. The refueling pool has the following subcompartments:

a. Upper guide structure laydown area

b. Core support barrel laydown area

c. Refueling area

The refueling pool filled with borated water forms a pool above the reactor vessel. The reactor vessel flange is sealed to the bottom of the refueling pool to prevent leakage of refueling water into the reactor cavity as described in Subsection 9.1.4.2.1.11. The fuel transfer tube connects the refueling pool to the refueling canal. The refueling pool is filled with borated water to a depth that limits the radiation at the surface of the water to
acceptable levels during the period when a fuel assembly is being transferred to the spent fuel pool. The shield walls that form the refueling pool are a minimum of 1.22 m (4 ft) thick.

The bottom of the refueling pool varies in elevation from 106 ft 6.375 in to 130 ft. The top of the refueling pool wall elevation is 156 ft. The bottom and wall inside of refueling pool are lined with 5 mm (3/8 in.) thick stainless steel plate.

3.8.3.1.8  **In-Containment Refueling Water Storage Tank**

The in-containment refueling water storage tank (IRWST) provides storage of refueling water, a single source of water for the safety injection and containment spray pumps, and a heat sink for discharges from the POSRVs and the reactor coolant gas vent system (RCGVS). The IRWST is annular and uses the lower section of the internal structure as its outer boundary. The IRWST is lined with a stainless steel liner plate to prevent leakage. The leak chase channels are seal welded to the back of the liner plate to control the potential leakage of borated water from the IRWST. The design features of the leak chase channel collection pipe drain system are described in Subsection 9.3.3.2.6. The IRWST consists of the top and bottom slab and the exterior wall. The bottom slab of IRWST rests on the reactor containment building basemat, and the top and bottom slabs are rigidly connected to the secondary shield wall. The IRWST is separated from the containment wall by a gap of 50 mm (2 in). The design of the IRWST considers pressurization as a result of the reactor containment building systems design basis accident. Refer to Section 6.8 for a description of the IRWST.

3.8.3.1.9  **Holdup Volume Tank**

The holdup volume tank (HVT) is a rectangular structural tank located between the primary shield wall and the IRWST inner wall. A screen is provided at the top of the HVT to prevent debris from getting into the tank. The HVT has a sump with pumps to measure the leakage rate and route the liquid to the liquid waste management system. During an accident, the water from breaks and the reactor containment building spray is collected in the HVT and overflows into the IRWST. Refer to Section 6.8 for a description of the HVT.

3.8.3.1.10  **Operating and Intermediate Floors**

The operating floor provides access for operating personnel functions and biological shielding. Intermediate floors provide access to equipment and components. The
operating floor is located at elevation 156 ft 0 in, and intermediate floors are located at elevations 114 ft 0 in and 136 ft 6 in. These floors consist of reinforced concrete or steel grating supported by structural steel framing that spans between the containment wall and the secondary shield wall. The steel framing has a horizontally sliding connection at the containment wall side to allow axial displacement of framing due to seismic displacement and thermal expansion. Openings are provided in the floor for equipment removal.

3.8.3.1.11 Interior Concrete Fill Slab

The interior concrete fill slab is located on the surface of liner plate of the reactor containment building basemat for protection of pressure boundary structures.

3.8.3.1.12 Polar Crane Supports

A large capacity of polar crane is supported by brackets installed in the containment shell, and the bracket is a steel structure consisting of cantilever beam.

3.8.3.2 Applicable Codes, Standards, and Specifications

The following codes, standards, and specifications are applied to the design of internal concrete and steel structures.

3.8.3.2.1 Design Codes and Standards

The design codes, standards, and regulations are listed in Table 3.8-1. The primary design code for concrete internal structure is ACI 349 (Reference 41) and ANSI/AISC N690 (Reference 42).

3.8.3.2.2 NRC Regulatory Guides

Conformance to each NRC RG is described in Section 1.9. The NRC RGs applicable to the design of the concrete and steel structures are 1.60, 1.61, 1.92, 1.122, 1.142, 1.199 (References 22 through 27), 1.69 (Reference 29), and 1.160 (Reference 43).

3.8.3.2.3 Industry Standards

Nationally recognized industry standards, such as those published by ASTM, will be used whenever possible to describe material properties, testing procedures, and fabrications and construction methods.
3.8.3.3 Loads and Load Combinations

The loads and load combinations for the CIS are obtained from Subsection 3.8.4.3.

The internal structures are designed for the following loads:

a. Dead load

b. Equipment operating loads and other live loads

c. Pipe break reactions

d. Seismic load (including 5% Accidental torsion)

e. Internal missiles (The internal structure is designed to withstand internal missiles, as described in Section 3.5. This includes pipe break missile.)

f. Pipe rupture jet impingement

g. Compartment accident pressure

h. The greatest pipe rupture loads from (1) pipe breaks not eliminated by leak-before-break, (2) the largest through-wall leakage crack in a high-energy line (minimum 37.9 L/min (10 gpm)), whether or not consideration of dynamic effects is eliminated by leak-before-break for the line, or (3) the largest leak from another leak source, such as a valve or pump seal.

i. Operating and accident temperatures

j. Hydrostatic and Hydrodynamic loads (POSRV load for IRWST)

k. Pipe, cable tray, duct, and ties

l. Pipe accident reaction

Miscellaneous loads \( (M_o, M_a) \), wind loads \( (W) \), precipitation loads \( (H, H_s) \), and tornado loads \( (W_t) \) of the loads in DCD Table 3.8-7A do not affect the design of the containment internal structures (CIS) because the CIS is protected by the concrete containment and there is no column, transformer, belt, etc. Therefore,
these loads are not applied to the design of the CIS. There is no large crane supported by the CIS. Jip cranes are evaluated in the local affected area.

Reactions of pipe, cable tray and duct ($R_o$) and pipe accident reactions ($R_a$) are evaluated in local affected area after the load informations are provided from the responsible design group and/or the supplier.

Flooding loads ($Y_f$) are not applied because it is offset by acting equally on both sides of the CIS. During operation, there is little to no pressure ($P_o$) acting on the CIS.

Seismic Category I concrete structures are designed for impulsive and impactive loads in accordance with the ACI 349 Code, and special provisions of Appendix C of the same code, with exceptions given in NRC RG 1.142. Impactive and impulsive loads are considered concurrent with seismic and other loads (i.e., dead and live loads) in determining the load resistance of structural elements.

Subcompartment pressure loads are the result of postulated high-energy pipe ruptures. In determining an appropriate equivalent static load for $Y_r$, $Y_j$, and $Y_m$, elasto-plastic behavior is acceptable with appropriate ductility ratios, provided excessive deflections do not result in loss of function of any safety-related system.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Analysis Procedure

The internal structure is designed for the loads and loading combinations specified in Subsection 3.8.3.3. The internal reinforced concrete structure including the reactor coolant system (RCS) is modeled with eight-node solid elements, three-node or four-node shell elements, and two-node beam elements using the ANSYS computer program. The design loads for analysis of internal structures are classified dead load, live load, hydrostatic and dynamic loads, temperature load, accident pressure load, pipe break load, and seismic load.

Dead loads include the self-weight of the PSW, SSW, IRWST, and fill concrete; equipment; and intermediate steel floor framing. Large equipment loads (e.g., reactor drain tank, letdown/regenerative heat exchanger, safety injection tank, recirculation fan) are treated as
dead loads. In addition, potential loads during construction or operating periods are treated as live loads.

The vertical and lateral pressures of liquids inside containment are treated as dead loads. Structures supporting fluid loads during normal operation and accident conditions are designed for the hydrostatic as well as hydrodynamic loads.

The thermal stress analysis is carried out by inputting the normal operating thermal load into the corresponding FEM of internal structure. During the thermal analysis, the equivalent uniform temperature gradient is input directly in the ANSYS model at the appropriate nodes. In the containment internal structure, the equivalent linear temperature profile for normal operating conditions is more severe than those of the accident conditions since the temperature difference between the inside and the outside surface of the containment internal structure during accident conditions is negligibly small. Therefore, the normal operating thermal load is conservatively applied for design of internal structures in case of normal operating and accident conditions.

The compartment pressures on internal structures are the result of a pipe break inside containment. In addition, branch line pipe break (BLPB) loads are dynamic reactions caused by the combined effects of branch line nozzle reactions or thrust due to pipe break, jet impingement on RCS equipment, or subcompartment pressure effects on RCS equipment. These loads are applied to the ANSYS model with pressure and concentrated loads.

Response spectrum analysis is performed for the two horizontal and one vertical seismic load directions using the methodology described in Subsection 3.7.2.

Regarding the analysis of subelements, the concrete slabs at elevations 114'-0", 136'-6", and 156'-0" between secondary shield wall and containment wall are analyzed using the FE (Finite Element) analysis program GTSTRUDL. A separate analysis model simulating each floor level is prepared and evaluated for each specified design load condition. To incorporate the proper seismic load on each slab, the response spectrum analyses are performed using the FRS which envelope containment shell side and secondary shield wall side at each elevation. In addition, the seismic anchor movement is considered in the seismic design case for concrete slabs.
3.8.3.4.2 Structural Design

The forces and moments resulting from the applied static and dynamic loads are used to design the walls, slabs, beams, and columns that make up the internal structure. The design and reinforcement are performed using ACI 349 or AISC N690 described in Subsection 3.8.4.4.

The walls and floors of internal structure are provided with anchor bolts for the mounting or attachment of RCS equipment. The jurisdictional boundary between supports and the building structure is defined by ASME Section III, Division 1, NF. According to the jurisdictional boundary of ASME Code, the component support and building structure is designed in accordance with ASME Section III, Division 1, NF and ACI 349 or ANSI/AISC N690, respectively. The anchorage for equipment supports to internal structure wall is provided in accordance with the requirements of ACI 349, Appendix B and NRC RG 1.199 described in Subsection 3.8.4.4.2.1. The details of the anchorage for RCS equipment are shown in Figures 3.8-19 through 3.8-22.

The design and analysis details for internal structures including critical sections are discussed in Subsection 3.8A.1.4.3.

3.8.3.4.3 Design Summary Report

A design summary report is prepared in Appendix 3.8A where the design of the representative critical sections of the structures is described.

The evaluation considering the deviations of as-procured or as-built construction to the design will be performed with the acceptance criteria described in Subsection 3.8.3.5.

3.8.3.5 Structural Acceptance Criteria

The structural acceptance criteria for the internal structures are outlined in Subsection 3.8.4.5.
3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Concrete Internal Structures

Materials, quality control, special construction techniques, and tolerances for the concrete internal structures are outlined in Subsection 3.8.4.6. The compressive strength of concrete is 6,000 psi at 91 days.

3.8.3.6.2 Structural Steel

The following materials are used:

a. Structural steel – ASTM A36, A572, A588, and A53
b. Bolts – ASTM A325, A490, and A307
c. Anchor bolts – ASTM 193, Grade B-7 and A36

Furnishing and fabrication of structural steel conform with all applicable requirements of AISC N690. Certified mill test reports for structural steel are submitted for review.

3.8.3.6.3 Stainless Steel Pool Liners

Stainless steel pool liners including IRWST liners and HVT liners are fabricated from ASTM A240, Type 304 material, hot rolled, annealed and pickled and further processed by cold rolling. This material quality is ensured in accordance with 10 CFR Part 50, Appendix B and ASME NQA-1.

Welding procedures are in accordance with ASME Section III, Division 2, CC-4540 and ASME Section IX. All seam welds are full-penetration butt welds. The liner plate seam welds are examined and tested as follows:

a. Liquid penetrant examination is performed on austenitic materials. The weld surfaces and at least 12.7 mm (1/2 in.) of the adjacent base material on each side of the weld are examined. The examination coverage is 100 percent of all shop and field seam welds.

b. Vacuum leak test is performed for leak-tightness on all liner plate seam welds.
Stainless steel surfaces on the pool sides of the plate conform with the finish of No.4 as specified by ASTM A480.

3.8.3.6.4 Stainless Steel Other Than Pool Liners

Stainless steel embedded plates and stainless steel checkered floor plates are fabricated from A240 Type 304 material, hot rolled, annealed, and pickled. Stainless steel bars and rounds are fabricated from A276 or A479 Type 304 material, hot rolled, annealed, and pickled.

Stainless steel pipes are fabricated from A312 Type 304 or A358 Type 304 or A376 Type 304 materials, hot rolled, annealed, and pickled.

Stainless steel gratings are fabricated from A240 Type 302 or Type 304 materials, hot rolled, annealed, and pickled prior to fabrication and then electro-polished after fabrication.

Stainless steel sump liners are fabricated from A240 Type 304 or Type 316 materials.

Stainless steel bolts are fabricated from A193 Type 304 class 1 material.

Stainless steel nuts are fabricated from A194 Type 304 material.

Stainless shapes are fabricated from A276 or A479 Type 304 materials.

3.8.3.7 Testing and Inservice Inspection Requirements

Testing and inservice inspection requirements are outlined in Subsection 3.8.4.7.

3.8.4 Other Seismic Category I Structures

The other seismic Category I structures of the APR1400 plant are the auxiliary building and the emergency diesel generator building.

The COL applicant is to provide the design of site-specific seismic Category I structures such as the essential service water building and the component cooling water heat exchanger building, essential service water conduits, component cooling water piping tunnel, and class 1E electrical duct runs (COL 3.8(5)).
3.8.4.1 Description of the Structures

3.8.4.1.1 Auxiliary Building

The auxiliary building houses the mechanical and electrical equipment used for normal plant operation and safe shutdown of the reactor. The auxiliary building is composed of the electrical and control area, main steam valve house, chemical and volume control system (CVCS) area, emergency diesel generator area, and fuel handling area.

The electrical and control area consists of the Class 1E electrical equipment rooms at elevation 78 ft 0 in and those areas located above them. The electrical and control area provides two physically separate divisions for electrical distribution, control, and instrumentation systems leading to the main control room (MCR). The upper floor of the electrical and control area contains the MCR, which is designed to provide security, fire, and environmental protection to the control equipment and the MCR operators.

The main steam valve house is a compartment located above the auxiliary feedwater (AFW) tank areas on the north and south sides of the auxiliary building. The compartment is from elevation 137 ft 6 in to 175 ft 0 in. The main steam valve house is designed to provide environmental protection, primarily missile protection, for the main steam and feedwater line safety-related valves and piping.

The CVCS area consists of a number of small rooms that are used to isolate components for water treatment required by operating systems. Individual rooms are used for radiation shielding.

The emergency diesel generator area provides protection to two diesel generators installed in separate compartments located on opposite sides of the auxiliary building.

The fuel handling area includes the spent fuel pool, refueling canal, cask loading pit, cask decontamination pit, truck/rail shipping bay, and new fuel storage area. The spent fuel pool is an open stainless steel lined reinforced concrete vessel used for submerged storage of radioactive spent fuel assemblies. The pool is approximately 10.8 m × 12.8 m (35 ft 6 in × 42 ft) with a depth of 12.8 m (42 ft). The walls and floor of the spent fuel pool are a minimum of 1.7 m (5 ft 6 in.) thick. The stainless steel liner plates cover the entire interior surface of the Fuel Handling Area. The wall liner and its anchorage system is designed and constructed to act initially as form work during concrete placement of walls, and subsequently as a leak tight membrane.
Fuel assemblies are transferred from the fuel handling area to the refueling pool via the refueling canal in the auxiliary building and then the fuel transfer tube in the reactor containment building. The refueling canal measures 1.8 m (6 ft) wide by 20.5 m (67 ft 3 in.) long. The minimum wall thickness on the fuel pool side is 1.8 m (6 ft). An opening in the fuel pool wall allows for passage of fuel between the fuel pool and the refueling canal. A steel divider is provided for the opening. Seals are incorporated to allow draining of the refueling canal while maintaining the water level in the spent fuel pool. An overhead bridge crane with a capacity of 150 tons is provided over the shipping bay and extending over the fuel pool and refueling canal. Interlocks are provided to prevent the crane from moving over the spent fuel storage area during cask handling operations. A new fuel-handling crane, running on rails mounted over the operating floor, is provided to handle the new fuel assemblies.

The two AFW tanks consist of three stainless steel lined reinforced concrete rooms. Each room has a single tank. The tanks extend from elevation 100 ft 0 in to the underside of the floor slab at elevation 137 ft 6 in.

The auxiliary building is rectangular with maximum dimensions of 106.0 m × 107.6 m (348 ft × 353 ft). It wraps around the reactor containment building with a seismic gap of 150 mm (6 in.). The auxiliary building shares common basemat structure with the reactor containment building. The auxiliary building is separated from other buildings by the isolation gap of 900 mm (3 ft).

There are no safety-related masonry walls in the Auxiliary Building.

The outlines of the auxiliary building are shown in Figures 1.2-9 through 1.2-19.

3.8.4.1.2 Emergency Diesel Generator Building

The emergency diesel generator (EDG) building block comprises two buildings, one that houses two additional generators and the other for the diesel fuel oil tank (DFOT). The two buildings are independent structures built on separate basements – one at elevation 100 ft 0 in for the EDG building, and the other at 63 ft 0 in for the DFOT building. The two basements are horizontally separated by an isolation gap of 900 mm (3 ft).

The EDG building block is a seismic Category I reinforced concrete rectangular structure. The EDG building is approximately 19.2 m (63 ft) wide, 40.8 m (134 ft) long, and 10.7 m (35 ft) high and the DFOT building is 20.3 m (66 ft 8 in.) wide, 21.6 m (71 ft) long, and
10.5 m (34 ft 6 in.) high. The EDG building block is separated from the other buildings by an isolation gap of 900 mm (3 ft).

The EDG and DFOT buildings are both single-story structures with a basemat and roof slab. The buildings are founded on a 1.2 m (4 ft) thick continuous mat foundation. The roof slab for the EDG building is at elevation 135 ft 0 in, and the roof slab for the DFOT building is at elevation 97 ft 6 in.

The lateral load-resisting system of the two buildings is composed of a diaphragm slab at roof level and shear walls monolithically interconnected at the roof. The lateral loads, such as wind load and horizontal earthquake load, are transferred to the soil foundation through the shear walls and the basemat. The vertical load-resisting system in each of the buildings consists of columns and shear walls. The vertical loads due to gravity and earthquake are carried by slabs, floor beams, columns, and shear walls down to the basemat and soil foundation.

There are no safety-related masonry walls in the EDG Building.

The outlines of the emergency diesel generator building are shown in Figures 1.2-20 and 1.2-22.

3.8.4.1.3 Fuel Storage Racks

The new fuel storage rack is bolted to embedments at the bottom of the new fuel storage pit and designed as a seismic Category I structure.

The spent fuel storage rack is designed as a free-standing type, i.e., neither anchored to the pool floor nor attached to the side walls, and designed as a seismic Category I structure. The new fuel storage rack and spent fuel storage rack are designed to meet the following criteria even under the plant abnormal condition, such as seismic or fuel handling accident:

a. Protect the stored fuel against a physical damage
b. Maintain the stored fuel in a subcritical configuration
c. Maintain the capability to load and unload fuel assemblies
d. Maintain the store fuel in a coolable geometry (spent fuel storage rack only)
The designs of the new fuel storage rack and spent fuel storage rack are described in Subsection 9.1.2.

3.8.4.1.4 Compound Building

The compound building is a non-safety-related seismic Category II structure with an embedment depth of approximately 12.80 m (42 ft). The compound building is separated from the auxiliary building by a 0.91 m (3 ft) gap. The top of basemat is at El. 63 ft 0 in. The exterior walls of the compound building are embedded 10.87 m (35 ft 8 in) with the finished grade at El. 98 ft 8 in. The compound building is rectangular and the major dimension is approximately 65.84 m (216 ft) long and 54.25 m (178 ft) wide, and 37.03 m (121 ft 6 in.) high.

The compound building is composed of reinforced concrete walls, columns, beams and slabs. The structural system of the compound building consists of 6 major floor slabs including the basemat and roof. The labyrinth walls that create numerous compartments utilized for the radwaste management system components are arranged on the basemat and first two floors. A bridge crane, supported below El. 139 ft 6 in on the east end of the building traverses the entire width of the crane area in the north-south direction.

The Class RW-IIa structures related to radwaste management of the compound building use ACI 349 and/or AISC N690, as applicable, in accordance with NRC RG 1.143. Also, the compound building shall be designed to preclude a structural failure that results from the SSE and that degrades the structural integrity of the adjacent auxiliary building.

The design codes, standards, specifications, regulations, regulatory guides, and other industry standards for all seismic Category I structures, other than the reactor containment building, described in Subsection 3.8.4.2 will be used for analysis and design of the compound building structures.

3.8.4.2 Applicable Codes, Standards, and Specifications

The following design codes, standards, specifications, regulations, Regulatory Guides, and other industry standards are used in the design, fabrication, construction, testing, and inspection of all seismic Category I structures other than the reactor containment building.
3.8.4.2.1  Design Codes and Standards

The design codes, standards, and regulations are listed in Table 3.8-1. The primary design codes for other seismic category I structures are ACI 349 (Reference 41) and ANSI/AISC N690 (Reference 42).

3.8.4.2.2  Regulatory Guides

The conformance of other seismic Category I structures to the applicable NRC RGs is addressed in Section 1.9. The NRC RGs that are applicable to the design of all seismic Category I structures other than the reactor containment building are NRC RGs 1.29 (Reference 28), 1.60, 1.61, 1.69 (Reference 29), 1.91 (Reference 30), 1.92, 1.115 (Reference 31), 1.122, 1.127 (Reference 44), 1.136 (Reference 45), 1.142, 1.143 (Reference 32), 1.199, 1.160 (Reference 43), and 1.221 (Reference 46).

3.8.4.2.3  Industry Standards

Nationally recognized industry standards, such as those published by ASTM, are used where practicable to define material properties, testing procedures, and fabrication and construction methods.

3.8.4.3  Loads and Load Combinations

This section presents the structural design load information for the APR1400 seismic Category I structures other than the reactor containment building. This load information consists of a summary list of major loads and load combinations. These load combinations are categorized on the basis of their nature, the probability of occurrence of each of the individual loads, and the probability of simultaneous occurrence of these loads to form a loading combination.

Evaluation of the capability of a structure for a given load combination is based on providing a factor of safety appropriate to the probability of occurrence. The appropriate factor of safety is reflected in the load factors and allowable stresses for the various load combinations.

The COL applicant is to evaluate any applicable site-specific loads such as explosive hazards in proximity to the site, projectiles and missiles generated from activities of nearby military installations, potential non-terrorism related aircraft crashes, and the effects of seiches, surges, waves, and tsunamis (COL 3.8(6)).
3.8.4.3.1 Normal Loads

a. Dead loads – (D)

Dead load refers to loads that are constant in magnitude and point of application. The types and definitions of dead loads and their combination requirements are given in Table 3.8-6.

b. Live loads – (L)

Live load refers to any normal loads that may vary with intensity and location of occurrence. The types and definitions of live loads and their combination requirements are given in Table 3.8-6. The specified design values for live loads are summarized in Table 3.8-5.

1) Soil and surcharge load (Lg)

Soil and surcharge load refers to load due to weight and pressure of soil, water in soil, or other material such as soil surcharge. These loads are applied on seismic Category I structures below grade, with the force directed toward the inside of the structures. Maximum elevation of groundwater is specified to be 0.61 m (2 ft) below plant grade for safety-related structures. For the construction loading condition, the minimum surcharge load is 48.0 kN/m² (1,000 psf) over any unoccupied area plus the actual construction loading surcharge from any known structures or load sources. For the normal loading condition, the minimum surcharge load is 24.0 kN/m² (500 psf). For the design of underground utilities, the minimum surcharge load for the construction loading condition is 24.0 kN/m² (500 psf) and for the normal loading condition is 12.0 kN/m² (250 psf). Any loading conditions that result in soil surcharge loads greater than these minimum values are checked on an individual basis.

2) Hydrostatic load (Lh)

Hydrostatic loads are due to weight and pressures of fluids with well-defined densities and controllable maximum heights or related internal moment and forces.

3) Snow load (Ls)
Based on the assumed site-related parameters, the 100-year snowpack roof load is considered to be 2.873 kN/m² (60 psf).

c. Thermal operating load \(- (T_o)\)

Thermal operating load is thermal load effect from the most critical transient or steady-state thermal condition at normal operation or shutdown conditions. This also includes thermal effects such as frictional loads due to expansion.

d. Equipment, pipe, cable tray, duct, and ties \(- (R_o)\)

This includes their dead load, live load, thermal load, seismic load, thrust load, and unbalanced internal pressure under construction, test, normal, severe, abnormal, and extreme environmental conditions.

\(R_{os}\) – Self-weights, including contents

\(R_{ot}\) – Transient or steady-state thermal loading conditions during normal operation and shutdown conditions

For the test loading condition, this includes piping reactions due to test cleanup and blowdown conditions.

\(R_{op}\) – Effects of unbalanced pressure and thrust

e. Crane and trolley loads \(- (C)\)

\(C\) is crane and trolley lifted load, including impact load, longitudinal load and lateral load. All of these loads are considered as acting simultaneously.

For the crane and trolley loads of the fuel handling overhead crane, the self-weight, lifting load, and vertical load due to vertical seismic acceleration are considered in the structural analysis of auxiliary building and NI common basemat at the worst location through 3 cases analyses. The lateral load due to seismic is not considered in the analysis due to the pendent effect.

1) Bridge crane

A bridge crane is a crane that has a bridge girder that moves longitudinally on two parallel beams, and a trolley hoist that moves laterally along the bridge
girder. It can either ride on a rail on top of the two parallel beams or hang from their bottom flanges.

The lifted load is the rated capacity of the main hook. The manufacturer usually provides the maximum wheel load to be applied to the support steel.

The vertical impact load is considered to be 25 percent of the maximum wheel loads.

The longitudinal load is considered as 10 percent of the maximum wheel loads of the crane applied at the supporting beam flange.

The lateral load on crane runways for bridge cranes (to provide for the effect of the moving crane trolleys) is considered as 20 percent of the sum of the lifted load and the crane trolley. The load is applied to each side at the supporting beam flange (top or bottom depending on the support arrangement) acting in either direction normal to the support beams, and is distributed based on lateral stiffness of the structure supporting the bridge crane.

2) Trolley

A trolley is a hoist that can move longitudinally but not laterally and is hung from the bottom flange of the support beam (trolley beam). It can rotate laterally. Trolleys may be motorized or hand-operated.

The lifted load is the rated capacity of the trolley hoist.

The vertical impact load is considered as 25 percent of the maximum wheel loads (lifted load plus the hoist self-weight).

The longitudinal load is considered as 10 percent of the maximum wheel loads of the trolley (lifted load plus the hoist self-weight), without impact, applied at the bottom flange of the support beam.

The lateral load for trolleys is calculated for the maximum wheel loads without impact, rotated a minimum angle of 5 degree about a vertical plane at the bottom flange. When an angle is specified on the equipment removal drawings and is greater than 5 degrees, it is used. The load is applied at the bottom flange of the support beam.
For severe environmental and extreme environmental loading combinations on seismic Category I structures, 100 percent of the lifted load is used.

f. Operating pressure – \( (P_o) \)

Operating pressure load is the external or internal air or gas pressure loads during normal operating conditions. Examples of this are pressures within air and gas ducts, and differential air pressures on building walls.

g. Miscellaneous normal loads – \( (M_o) \)

\( M_o \) is other miscellaneous normal loads, such as the column contingency load, transformer high-voltage line pull-off loads, vehicle loads, miscellaneous material handling loads (e.g., belt pulls, barge berthing impact). Column contingency loads are considered in the column design to account additional loads.

h. Stability load (S)

Stability load (S) includes overall (Frame) column stability loads (SC) and girder stability loads (SG). Overall stability loads may not need to be considered if the effects of wind or seismic loads on the structure are significant. Stability loads are only used in the design of horizontal bracing, moment frames, vertical bracing diagonals, and collector elements (beams or struts which transfer loads to diaphragms, horizontal bracing, vertical bracing, shear walls or moment frames). Additional load combinations shall be utilized as necessary to delete stability loads from the design of other structural elements, such as columns and anchor bolts. P-D analyses are not required for stability loads.

i. Construction loads

Construction loads are related to all events and loads during construction. These include dead loads, live loads, temperature loads, precipitation loads, wind loads, and construction loads such as surcharge loads due to construction equipment, hoisting loads due to construction activities, and laydown loads during construction. SEI/ASCE 37-02 (Reference 8) is considered to be supplemental guidance.
3.8.4.3.2 Abnormal Loads

a. Accident pressure – \( P_a \)

Accident pressure is applied to external or internal air, gas, or liquid pressure loads during abnormal operating conditions. Examples of this are excursion pressures within gas ducts due to fan or damper type failures and differential air pressure on a building wall due to a postulated pipe break including annulus pressurization effects and flooding loads. An appropriate dynamic factor to account for the dynamic response of the structure and the time dependency of the load is included.

1) Main steam valve house

The compartmental accident pressures due to main steam and feedwater line breaks are considered.

2) Other areas

Accident pressures in other areas of seismic Category I structures are defined during plant layout and design.

b. Accident temperature – \( T_a \)

Accident temperature is thermal load effects during abnormal operating conditions. Accident temperatures in other areas of seismic Category I structures are defined during plant layout and design.

c. Accident reactions of equipment, pipe, cable tray, duct, and ties – \( R_a \)

\( R_a \) is reactions of pipe, cable tray, duct, and ties. This includes their dead load, live load, thermal load, seismic load, thrust load, and transient unbalanced internal pressure loads under abnormal and/or extreme environmental conditions.

d. Pipe break reactions – \( Y_r \)

\( Y_r \) is equivalent static load on the structure generated by the reaction of the broken high-energy pipe during the postulated break, including an appropriate dynamic factor to account for the dynamic response of the structure, and the time dependency of the load.
1) Pipe whip restraint reactions

Pipe whip restraint reactions are the loads transferred from the restraint to the supporting structure.

2) Pipe hanger loads

This is the portion of the pipe hanger reaction that is due to a pipe break (excluding thermal effects).

e. Jet impingement load – \( Y_j \)

\( Y_j \) is the jet impingement equivalent static load on a structure generated by a postulated pipe break, including an appropriate dynamic factor to account for the dynamic response of the structure, and the time dependency of the load.

f. Missile impact load – \( Y_m \)

\( Y_m \) is the missile impact load on a structure generated by or during a postulated pipe break, like pipe whipping, including an appropriate dynamic load factor to account for the dynamic response of the structure, and the time dependency of the load.

g. Flooding load – \( Y_f \)

\( Y_f \) is the load within or across a compartment and/or building due to flooding generated by a postulated pipe break. These loads are calculated considering the design basis flood heights.

h. Miscellaneous abnormal loads – \( M_a \)

\( M_a \) includes other miscellaneous site-related accidents such as blast, aircraft impact, or internally generated equipment missiles.

i. POSRV load \( (P_{ac}) \)

\( P_{ac} \) is air-clearing load, which is the hydrodynamic load generated by the expulsion of air in POSRV discharge lines during the POSRV discharge following the water clearing phenomena in the sparger.
3.8.4.3.3 Severe Environmental Loads

a. Wind loads – (W)

W is the equivalent static load generated by the design wind velocity, and is calculated in accordance with ASCE 7 (Reference 33) and described in Subsection 3.3.1. Seismic Category I structures are designed for a 100-year recurrence interval wind, and for tornado and hurricane winds and missiles as described in Subsections 3.3.2 and 3.5.1.4.

For seismic Category I structures, an importance factor I of 1.15 is used with 50-year, 3-second gust speed at exposure Category C, as defined in ASCE 7, Section 6.5.

b. Design flood/precipitation – (H)

Flood loads on seismic Category I structures are determined based on the maximum site flood levels specified in Chapter 2.

c. Operating basis earthquake

The operating basis earthquake (OBE) is defined as one-third of the SSE. Therefore, an analysis or design of APR1400 seismic Category I SSCs based on OBE is not required in accordance with Appendix S of 10 CFR Part 50.

3.8.4.3.4 Extreme Environmental Loads

a. Safe shutdown earthquake – (E_s)

SSE loads are considered as follows:

1) Seismic Category I structures

For seismic Category I structures, E_s are the loads generated by the SSE. Hydrodynamic load and dynamic soil pressure are included in E_s. The hydrodynamic load and dynamic soil pressure are added to the SRSS-combined seismic load by the absolute sum.

Seismic response for SSE is determined using a dynamic analysis. Enveloped floor response spectra in both the horizontal (N-S and E-W) and
vertical directions are prepared for all major building floor elevations. An equivalent static method is used to determine SSE loads on structural components (e.g., floor slabs, beams).

2) Combination of SSE loads

For each load, the responses from all three directional earthquakes are combined simultaneously. The independent directional responses are combined using the square root of the sum of the squares (SRSS) method or the 100-40-40 percent rule described in ASCE 4, Subsection 3.2.7. For the auxiliary building and the EDG building, the square root of sum of the square method was used to combine the SSE loads.

Stresses due to seismic loads from different directions are combined by the SRSS method using the following expression:

\[ R = \pm \sqrt{\sum_{i} R_i^2} \]

Where R is any response of interest and R_i (i=1, 2, 3) is the two horizontal components and one vertical component of earthquake motion.

3) Additional seismic loads due to accidental torsion

Additional seismic loads due to accidental torsion are accounted for as required by SRP 3.7.2, II.11. An additional eccentricity of the mass at each floor equivalent to 5 percent of the maximum building dimension is included. The accidental torsion load is represented by an additional shear force at each floor elevation determined from the analysis for the product of resultant story shear and accidental eccentricity at each elevation.

b. Tornado or hurricane load – (W_t)

The tornado or hurricane loads are described in Subsection 3.3.2.

c. Probable maximum flood/precipitation – (H_s)
Hs is the forces, due to the probable maximum precipitation as well as the maximum flood level, which includes the effects of seiches, surges, waves, and tsunamis.

3.8.4.3.5 Other Loads

Other loads are loads resulting from aircraft hazard and explosion pressure wave that are not included in the design basis. These loads are evaluated to prevent damage to safety-related structures, systems, and components beyond the design basis condition.

3.8.4.3.6 Load Combinations

The load combinations to be used in the design of the structure are in accordance with Tables 3.8-7A and 3.8-7B, and in conjunction with the definitions of load conditions and design loads as provided in Subsections 3.8.4.3.1 through 3.8.4.3.5.

3.8.4.3.7 Below Grade Exterior Walls

The design and analysis procedures for seismic Category I exterior walls below grade are described below.

Hydrostatic (groundwater)

The hydrostatic unit water pressure (Pw) at a depth h below ground level is calculated as a linearly distributed pressure depending on the design water level (0.61 m (2 ft) below plant grade).

\[ P_w = \gamma_w h \]

Where:

\[ \gamma_w = \text{unit weight of water} = 62.4 \text{ pcf} \]

Static Earth Pressure

Static earth pressure is based on “at-rest” conditions and the coefficient of earth pressure is calculated as the following relationship. In addition, the soil parameters are described in DCD Table 2.0-1.

\[ P_s = K_o \gamma_h \]
Where:

\[ K_o = 1 - \sin(\phi) = \text{Coefficient of earth pressure at rest condition} \]

\[ \gamma = \gamma_s = \text{Soil density in saturated condition} \]

\[ \gamma = \gamma_{sub} = \text{Soil density in submerged condition} \]

**Surcharge Pressures**

The surcharge pressure is defined for all soil cases as the at-rest pressure as follows:

\[ P_{sur} = K_o q \]

Where:

\[ q = \text{static surcharge pressure} \]

The dynamic lateral surcharge pressure is the same as the static surcharge pressure, for conservatism.

**Dynamic Earth Pressures**

The dynamic earth pressure is calculated in accordance with ASCE 4 (Reference 47), Section 3.5.3, Figure 3.5-1, “Variation of Normal Dynamic Soil Pressures for the Elastic Solution.”

**Dynamic Groundwater Pressures**

Dynamic groundwater pressure is calculated based on the hydro-dynamic formula suggested by Matsuo and O’Hara in “Principles of Soil Dynamics,” written by Braja M. DAS (Reference 48). The design water level (0.61 m (2 ft) below plant grade) is considered in the calculation of hydrodynamic water pressure.

**Passive Earth Pressure**

The passive earth pressure is not included in the resistance force for sliding and overturning in the basemat stability check. Therefore, passive earth pressure on exterior walls does not need to be considered.
3.8.4.4 Design and Analysis Procedures

The auxiliary building and the emergency diesel generator building are composed of basemat foundation, rectangular walls, floor slabs, columns, and beams. The slabs and shearwalls in the building represent the primary lateral and vertical load-resisting system and are designed for both gravity- and seismic-related loads. Concrete slabs at various elevations in the building distribute lateral forces (via diaphragm action) to the shearwalls as in-plane loads and resist vertical forces (self-weight and seismic forces) as out-of-plane loads. Lateral loads are transferred down to the basemat foundation through shearwalls as in-plane shear forces and moments. Vertical loads on slabs are supported by concrete beams or walls. The loads are transferred to the basemat foundation by the walls and the frames composed of concrete beams and concrete columns. In addition to the structural components, components are designed to provide biological shielding and protection against tornado, hurricane, and turbine missiles.

Structural analyses of the concrete structures are performed by the ANSYS (Reference 9) or GTSTRUDL (Reference 34) program to determine their design forces due to various loads and load combinations.

Other seismic Category I concrete structures are analyzed and designed in accordance with the requirements of ACI 349 with exceptions of the requirements in NRC RG 1.142. Those requirements are incorporated into the design and accommodated in the load combinations described in Subsection 3.8.4.3 for concrete structures.

The design and analysis details for AB and EDGB, including the critical section, are discussed in Subsection 3.8A.2.4 and 3.8A.3.4, respectively.

Other seismic Category I steel structures are designed in accordance with AISC N690 using the allowable stress design method.

The new fuel storage rack in the new fuel storage pit and spent fuel storage rack in the spent fuel pool are designed to withstand the seismic load that is seismic excitation along three (3) orthogonal directions is applied simultaneously for the design of the rack.

The designs of the new fuel storage rack and spent fuel storage rack are described in Subsection 9.1.2.
The new fuel storage pit is covered by steel plates. The steel plate is designed not to fall or collapse for the protection of new fuel and new fuel storage racks in the event of an SSE in accordance with the requirements in AISC N690. The COL applicant is to perform the analysis and design of the steel plate for the new fuel storage pit (COL 3.8(7)).

**3.8.4.4.1 Analysis of Structure**

A detailed three-dimensional finite element model of the other seismic Category I structures is developed to distribute the global loads to all structural components. The structural analysis model of the auxiliary building is shown in Figure 3.8-23. The model includes all walls, floor slabs, and major structural beams and columns. The walls and slabs are modeled with three- or four-node shell elements, and the beams and columns are modeled with two-node beam elements using the ANSYS computer program.

Global static analysis for each loading condition is performed using the models, and results are combined using the load combinations identified in Subsection 3.8.4.3.4. Seismic analyses of seismic Category I structures conform with the procedures described in Subsection 3.7.2. The accelerations from the seismic analysis are applied to the finite element models as equivalent static loads at the corresponding elevations.

**3.8.4.4.2 Structure Design**

**3.8.4.4.2.1 Concrete Structure**

The requirements for the design of seismic Category I concrete structures conform with all requirements of ACI 349 and NRC RG 1.142.

Design provisions for impulsive and impactive effects in the seismic Category I structures are in accordance with ACI 349, Appendix C.

ACI 349Appendix A, ACI Report ACI 349.1R or computer analysis programs are used to evaluate thermally induced forces and moments in seismic Category I structural members.

Required reinforcing for the seismic Category I concrete members are determined in accordance with applicable code provision as follows:

**In-Plane Shear**

Design for in-plane shear is in accordance with the requirements of ACI 349, Chapter 11.
Out-of-Plane Shear

Design for out-of-plane shear is in accordance with the requirements of ACI 349, Chapter 11.

Bending and Axial Loads

Design for out-of-plane bending with axial compression (or tension) is in accordance with the requirements of ACI 349, Chapter 10.

Minimum reinforcement of design section is in accordance with the requirements of ACI 349, Chapter 7 and 10.

Design for lateral load resisting system, concrete frame, concrete beam and concrete column is in accordance with the requirements of ACI 349, Section 21.5 through 21.7.

When feasible, uniform reinforcement patterns are used for sections with similar requirements, thickness, and loading.

The design of the support anchorage to the concrete structure meets the requirements of ACI 349-01 Appendix B and NRC RG 1.199.

3.8.4.4.2.2 Steel Structure

The design of seismic Category I steel structures and components uses the allowable stress design methods in accordance with AISC N690 including Supplement 2.

Bolted connections are used for field erection of structural steel beams and columns. The design of bolted connections is in accordance with Section Q1.16 of AISC N690 and the “Specification for Structural Joints Using ASTM A325 or A490 Bolts” (Reference 35).

Welding activities associated with seismic Category I structural steel and their connections are accomplished in accordance with the requirements of AWS D1.1 (Reference 36).

3.8.4.4.2.3 Missile Protection

Exterior walls and roof slabs of seismic Category I structures function as missile barriers. Design of missile barriers provides reasonable assurance that the structure will not collapse under the missile load and the barrier will not be penetrated. Safety-related SSCs are
protected from secondary missiles as a result of backface scabbing. Interior walls and floors are designed to function as missile barriers when it is evaluated to be necessary.

The design of seismic Category I structures for internally generated and externally generated missiles conforms with the procedures described in Section 3.5.

3.8.4.4.2.4 **Flooding**

Flooding is addressed in Section 3.4.

3.8.4.4.2.5 **Wall/Floor Penetrations**

Openings are acceptable without analysis if they meet the criteria in ACI 349, Section 13.5.2.

Penetration sleeves usually consist of a pipe embedded in a concrete wall or concrete floor with a short projection at one of both faces. As a minimum, penetration sleeves have sufficient thickness to maintain roundness during concrete pouring of other construction. Penetration sleeves are designed in accordance with ACI 349 and AISC N690.

Each corner of rectangular openings in walls or slabs is provided with additional reinforcing to reduce cracking due to stress concentration at these locations in accordance with ACI 349, Section 14.3.7.

3.8.4.4.2.6 **Embedment Plates**

Embedment plates are located throughout the plant to provide sufficient and efficient support for the various structures and components. The plate is designed in accordance with AISC N690. The anchorage to concrete is designed in accordance with ACI 349-97, including Appendix B (2001), and NRC RG 1.199.

3.8.4.4.2.7 **Stainless Steel Liner Design**

The design of stainless steel liners is similar to the containment liner design as described in Subsection 3.8.1.4.10. The allowable stresses and strains for the liner are consistent with ASME Section III, Division 2, CC-3000.
3.8.4.4.3 Design Summary Report

A design summary report is prepared for the other seismic Category I structures in Appendix 3.8A where the design summaries for the representative critical sections of the structures are described.

Deviations from the standard design of as-procured or as-built construction are acceptable based on an evaluation following the methods and procedures described in Sections 3.7 and 3.8. The structural design is evaluated in accordance with the acceptance criteria described in Subsection 3.8.4.

3.8.4.5 Structural Acceptance Criteria

Structural acceptance criteria for design strengths and allowable stresses are listed in Table 3.8-7A for concrete structures and in Table 3.8-7B for steel structures, and are in accordance with ACI 349 and AISC N690, except as provided in the table notes.

Shear reinforcement is provided in accordance with ACI 349, Chapter 11.

Reinforcement for bending with axial compression (or tension) is provided in accordance with ACI 349, Chapter 7 and 10.

Reinforcement for lateral load-resisting systems, concrete frames, concrete beams, and concrete columns is provided in accordance with ACI 349, Section 21.5 through 21.7.

Limits for crack controls, deflections, and other design criteria are in accordance with ACI 349 and NRC RG 1.142 for concrete structures and AISC N690 for steel structures.

The structural acceptance criteria for the foundation design are described in Subsection 3.8.5.5.

The seismic Category I concrete structures that are subjected to thermal effects conform with the minimum provisions of ACI 349 Appendix A.

The structural acceptance criterion on the new fuel storage rack and spent fuel storage rack is to meet the maximum allowable stress limits with given load combinations described in Table 3.8-7C in accordance with the SRP 3.8.4, Appendix D. When the effects of seismic loads are considered, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions is in accordance with the SRP 3.8.5, II.5.
3.8.4.6 Material, Quality Control, and Special Construction Techniques

This section contains information relating to the materials, quality control programs, and special construction techniques and tolerances used in the fabrication and construction of the seismic Category I concrete and steel structures other than the reactor containment building.

The tolerances for structural concrete shall be in accordance with ACI 117, and for structural steel shall be in accordance with ANSI/AISC 303. Any provisions in these codes which provide acceptance criteria for conditions when tolerances are exceeded shall not apply, but a licensee referencing the APR1400 DCD may deviate from the tolerances in these codes using alternative acceptance criteria, if these acceptance criteria are approved by the NRC. As-built dimensions will also be evaluated to verify compliance with the design bases and applicable codes and standards identified in Table 3.8-1. The COL applicant may provide construction tolerance acceptance criteria and the basis for the criteria (e.g., through the use of analysis, industry research, or testing) for cases where the tolerances in the ACI 117 and ANSI/AISC 303, for structural concrete and structural steel, respectively, may be exceeded (COL 3.8 (22)).

3.8.4.6.1 Material

The seismic Category I structures are poured-in-place reinforced concrete structures. The major materials that are used in the construction are concrete, reinforcing bars, and structural steel.

3.8.4.6.1.1 Concrete

The minimum concrete compressive strength used in other seismic Category I structures is 34.5 MPa (5,000 psi) at 91 days. The basic ingredients of concrete are cement, fine aggregates, coarse aggregates, and mixing water. Admixtures may be used if needed. The concrete conforms with ACI 349 and ASTM C94.

The COL applicant is to determine the environmental condition associated with the durability of concrete structures and provide the concrete mix design to prevent concrete degradation caused by factors such as the reactions of sulfate and other chemicals, the corrosion of reinforcing bars, and the effect of reactive aggregates (COL 3.8(8)).
Cement is Type I and conforms with ASTM C150. In special circumstances, other approved cements may be used.

Aggregates conform with ASTM C33.

The water and ice used in mixing concrete are clean and free from injurious amounts of oils, acids, alkalis, salts, organic materials, or other substances that may be deleterious to concrete or steel. The water and ice do not contain more than 500 ppm of chlorides as Cl⁻, as determined in accordance with ASTM D512 and not more than 2,000 ppm of total solids as determined in accordance with ASTM D1888. A comparison of the proposed mixing water properties is made with distilled water by performing the following tests:

a. Time of setting, in accordance with ASTM C191 (Reference 37). The results obtained for the proposed mixing water conform with ASTM C 94.

b. Compressive strength, in accordance with ASTM C109 (Reference 38). The results obtained for the proposed mixing water are not lower by more than 10 percent of those obtained for distilled water.

The water used to make ice for concrete pours in hot weather conforms with the requirements for mixing water described above.

Admixtures, if used and as determined by detailed mix design, conform with the applicable ASTM standards, as follows:


The ingredient materials are stored in accordance with the recommendations in ACI 304R.

Concrete mixes are designed in accordance with ACI 301. The batching, mixing, and transporting of concrete conform with ACI 301. The placement of concrete, consisting of preparation before placing, conveying, depositing, protection, and bonding is in accordance with ACI 301.

3.8.4.6.1.2 Reinforcing Steel

Reinforcing steel consists of deformed reinforcing bars conforming to ASTM A615, Grade 60, or ASTM A706, Grade 60. The fabrication of reinforcing bars, including fabrication tolerances, is in accordance with ACI 315 in ACI Detailing Manual (SP-66) (Reference 49). The placing of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars and field tolerances is in accordance with ACI 349. Epoxy-coated reinforcing steel may be used for areas where a corrosive environment is encountered.

3.8.4.6.1.3 Structural Steel

Structural steels are used as follows:

Other structural steels listed in AISC N690 may also be used.


Fabrication and erection of structural steel in seismic Category I structures are in accordance with the requirements of AISC N690.
Welding materials conform with the requirements of the Structural Welding Code (AWS-D1.1). AWS D1.1 Table 3.1 shows the compatibility of filler metal with base metal. AISC N690 provides supplemental information on weld materials for stainless steel.

Bolted connections conform with one of the following specifications:


b. ASTM A490, “Standard Specification for Heat-Treated Steel Structural Bolts, 150 ksi Minimum Tensile Strength”

c. ASTM A307, “Standard Specification for Carbon Steel Bolts and Studs, 60,000 psi Tensile Strength”

Bolts listed in AISC N690 may also be used.

3.8.4.6.1.4 Stainless Steel

Stainless steel pool liners are fabricated from ASTM A240, Type 304 material, hot rolled, annealed and pickled and further processed by cold rolling. This material quality is ensured in accordance with 10 CFR Part 50, Appendix B and ASME NQA-1. Further requirements for stainless steel pool liners and other stainless steel are described in Subsections 3.8.3.6.3 and 3.8.3.6.4. Welding material for the stainless steel pool liners is used in accordance with ASME Section III, CC-2600 and ASME Section IX.

3.8.4.6.2 Quality Control

The quality of materials is controlled by requiring the suppliers to furnish appropriate mill test reports as required under relevant ASTM specifications as described in Subsection 3.8.4.6.1. The mill test reports are reviewed and approved in accordance with the general provisions of the overall quality assurance program outlined in Chapter 17 and supplemented by the special provisions of the appropriate codes and specifications for design listed in Subsection 3.8.4.2.

Erection tolerances, in general, are in accordance with the referenced design code. Where special tolerances that influence the erection of equipment are required, they are indicated on the design drawings.
3.8.4.6.3 Special Construction Techniques

No special construction techniques are used in the construction of other seismic Category I structures.

The corrosion protection of the auxiliary building reinforcing steel is provided by an adequate cover of high-quality concrete over the reinforcing bar. Since the APR1400 concrete structures are built with concrete of maximum water-cementitious material (w/cm) ratio of 0.4 and minimum compressive strength (f'c) of 5,000 psi, the auxiliary building concrete structures will be prevented from penetration of chloride or sulfide ions based on the ACI 349 Section 3. Unless the concrete is penetrated by chloride or sulfide ions, the reinforcing bar remains passive and will not corrode.

Slabs in the auxiliary building are constructed using metal deck and steel beams that support metal deck and concrete slab during construction. Steel beams are connected to shear walls or concrete beams. This method allows concrete slabs to be constructed without shorings and forms.

There are two methods to install stainless steel liner plates (SSLP) in spent fuel pool, refueling canal, cask loading pit and cask decontamination pit:

a. Wall-paper type (floors) liner plates are field welded to stainless steel liner embedment strips in the concrete.

b. Form type (wall) liner plates (with its anchorage system) are field welded together as a complete unit or shop welded together as a module at the SSLP assembly filed shop which act initially as form work during concrete placement of walls and subsequently as a leak tight membrane.

The COL applicant is to determine construction techniques to minimize the effects of thermal expansion and contraction due to hydration heat, which could result in cracking (COL 3.8(9)).

3.8.4.7 Testing and Inservice Inspection Requirements

There is no testing or in-service surveillance beyond the quality control tests performed during construction, which is in accordance with ACI 349, AISC N690, or ANSI N45.2.5, in accordance with NRC RG 1.127 and NUMARC 93-01.
For other seismic Category I structures outside containment, the structures monitoring and maintenance requirements program is to be in accordance with 10 CFR 50.65 and NRC RG 1.160.

The structures are monitored in accordance with Paragraph (a)(2) of 10 CFR 50.65, provided there is not significant degradation of the structures. The condition of all structures is assessed periodically. The appropriate frequency of the assessments is commensurate with the safety significance of the structures and their condition.

For water control structures, the inservice inspection program is to be in accordance with NRC RG 1.127. Water control structures covered by this program include concrete structures, embankment structures, reservoirs, cooling water channels and canals, intake and discharge structures, and safety and performance instrumentation.

It is important to accommodate inservice inspection of critical areas. Monitoring and maintaining the condition of the other seismic Category I structures is essential for plant safety. Special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high-radiation areas) to accommodate inservice inspection of other seismic Category I structures is provided on a case-by-case basis.

For plants with nonaggressive ground water/soil, (i.e., pH > 5.5, chlorides < 500 ppm, sulfate < 1,500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to examine the exposed portion of below-grade concrete, when excavated for any reason, for signs of degradation, and to conduct periodic site monitoring of ground water chemistry to confirm that the ground water remains non aggressive.

For plants with aggressive ground water/soil, (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a surveillance program to monitor the condition of normally inaccessible, below-grade concrete for sign of degradation.

Therefore, for safety and serviceability of seismic Category I structures during the operation of the plant, the COL applicant is to provide appropriate testing and inservice inspection programs to examine the condition of normally inaccessible, below-grade concrete for signs of degradation and to conduct periodic site monitoring of ground water chemistry. Inservice inspection of the accessible portion of concrete structures is also to be performed. (COL 3.8(10)).
Monitoring of seismic Category I structures is performed in accordance with NRC RG 1.160.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The foundation basemat is a reinforced concrete common basemat structure for the nuclear island that consists of the reactor containment building and auxiliary building.

Waterproofing membranes are used for exterior horizontal and vertical surfaces of all seismic Category I building structures. The horizontal waterproofing membranes are installed between lower and upper lean concrete beneath the basemat as shown in Figure 3.8-27. The COL applicant is to verify that the coefficient of friction between the lean concrete and waterproofing membrane is greater than or equal to 0.55 (COL 3.8(11)).

3.8.5.1.1 Reactor Containment Building Foundation

The reactor containment building basemat is reinforced at the top and bottom with layers of reinforcing steel bars. The reinforcing bars are arranged in radial and hoop directions for top layers and orthogonal directions for bottom layers. In addition, the reinforcing bars at the floor of the reactor pit below the liner are arranged in orthogonal directions for the top and bottom layers.

The steel liner plate for the containment basemat is 6.0 mm (0.25 in.) thick except for embedments in local areas where it is thickened. The liner is anchored by welding on the top of the structural steel rolled sections embedded in the concrete.

Interior structural concrete is poured over the basemat liner to provide support for the reactor coolant loop (RCL) equipment, RCL piping, and the interior concrete walls. Tensile loads generated from analyses are carried by anchorage through the liner plate and into the basemat, if required. Tensile loads from internal concrete walls are transferred from the wall reinforcement to a thickened liner plate using mechanical splices and are then transferred to the base slab through steel reinforcement dowels welded to the underside of the thickened liner plate.
3.8.5.1.2  **Auxiliary Building Foundation**

The foundation of the auxiliary building is a reinforced concrete mat and rests on competent material with a thickness of 3.05 m (10 ft). The bottom of the basemat is located at elevation 40 ft 0 in and 45 ft 0 in, below the finished grade elevation.

The auxiliary building basemat is reinforced at the top and bottom with layers of reinforcing steel bars. The reinforcing bars are arranged in orthogonal directions for the top and bottom layers.

3.8.5.1.3  **Emergency Diesel Generator Foundations**

The emergency diesel generator (EDG) building block comprises two buildings, one of which houses the EDGs and the other the diesel fuel oil tank (DFOT). The two buildings are independent structures built on a separate concrete reinforced mat foundation with a thickness of 1.2 m (4 ft). The bottom of the basemat is located at elevation 92 ft 0 in for the EDG building and elevation 59 ft 0 in for the DFOT building.

3.8.5.2  **Applicable Codes, Standards, and Specifications**

The reinforced concrete foundations of the reactor containment building are designed using the codes and standards described in Subsection 3.8.1.2. The reinforced concrete foundations and supports of other seismic Category I structures are designed using the codes and standards described in Subsection 3.8.4.2.

3.8.5.3  **Loads and Load Combinations**

The design loads and load combinations of the reactor containment building foundation are described in Subsection 3.8.1.3 and Table 3.8-2. The design loads and load combinations of the auxiliary building foundation and EDG building foundation are described in Subsection 3.8.4.3 and Table 3.8-7A. For combining SSE loads, the 100-40-40 percent rule was used for the basemat analysis to combine independent directional loads from superstructures, in accordance with ASCE 4, Subsection 3.2.7. The 100-40-40 percent rule is based on the observation that the maximum increase in the resultant for two orthogonal forces occurs when these forces are equal. This rule may be applied for combining responses in the same direction due to different components of motion.

The analysis and design of the foundation below the reactor containment building conform to the requirements of ASME Section III, Division 2, CC. The analysis and design code
for the foundations below the auxiliary building and EDG building is ACI 349. As the
design criteria are different, the applications of loads and load combinations for the
foundations are in accordance with each code. While the ACI 349 code concentrates on
the requirements as one of the concrete structures in a nuclear power plant, the ASME Code
describes the requirements with more focus on the functionality of containment. The load
factors in Tables 3.8-2 and 3.8-7A are based on these design concepts according to the two
codes. For the code application scope and jurisdiction boundary of the NI common
basemat, refer to Subsection 3.8.1.1.2 and Figure 3.8-26.

3.8.5.4 Design and Analysis Procedures

The NI common basemat is analyzed using the ANSYS computer program. Stiffening
effects of the reactor containment building wall, internal concrete structures, and auxiliary
building are included in the model.

The NI common basemat is modeled with eight-node solid element in the ANSYS
computer program. In addition, in order to consider the soil effect, the link element in
ANSYS is used with the NI common basemat model for static loading case. In case of
seismic loading, the foundation model is used with the NI common basemat model.
Detailed explanation of spring and foundation stiffness is described in Subsection 2 of

The reinforced concrete basemat of the reactor containment building is designed in
accordance with ASME Section III, Division 2, CC. Other seismic Category I basemats of
reinforced concrete are designed in accordance with ACI 349 and the provisions of NRC
RG 1.142 where applicable.

As the design criteria for the RCB and AB area of the NI common basemat are different,
the application of loads is also divided into two parts, as shown in Figure 3.8-26. The load
combinations provided by the ASME and the ACI Codes are used in the analysis and design
of the RCB and AB foundations, respectively. Regarding the portion beyond the RCB
foundation directly beneath the containment shell, the following aspects are additionally
considered in the analysis and design of NI common basemat.

At the interface between the two codes, a greater amount of reinforcement required by
either code is used, and the reinforcement of the RCB foundation is developed into the AB
foundation as shown in Figures 3.8A-16 and 3.8A-17. The provisions of both codes are
used to select a conservative development length.
The outside portion of the RCB foundation, i.e., the entire AB foundation area, is conservatively designed using the larger member forces from the analysis results of the ASME and ACI Codes.

For the effect of the ACI load combination on the RCB foundation, the load combinations in the ASME and the ACI Codes are compared. Four kinds of loads that exist in the ACI load combination but do not exist in the ASME load combination are investigated: operating pressure, miscellaneous loads, crane and trolley load, and hydrostatic load. The operating pressure and miscellaneous loads do not have an effect on the global behavior of the basemat, and these loads are not considered in the NI common basemat analysis. The crane and trolley load is applied in the analysis as self-weight of the fuel handling overhead crane. The hydrostatic load of water storage tanks in the AB has minor effect on the RCB foundation.

In addition, the effect of the soil pressure on the RCB foundation is negligibly small because the NI common basemat has a big dimension of 106.0 m × 107.6 m (348 ft × 353 ft) and the distance between the RCB foundation and AB outside wall is accordingly long. Most of the soil pressure on the side walls of the AB foundation is transmitted through the AB internal walls and has little effect on the behavior of the RCB foundation.

The design and analysis details for the foundations of safety-related structures are discussed in Subsections 3.8A.1.4.2, 3.8A.2.4.1 and 3.8A.3.4.1.

The maximum allowable differential settlement of foundation is 12.7 mm per 15.24 m (0.5 in per 50 ft) within NI common basemat. The maximum allowable differential settlement between buildings is provided in Table 2.0-1 based on enveloping properties of subsurface materials. In addition, the common basemat is analyzed for construction sequences to minimize any potential differential settlement during construction.

3.8.5.4.1 Analyses for Loads during Operation

The reinforced concrete foundations of seismic Category I structures are analyzed and designed for the reactions due to static, seismic and all other loads that affect basemat analysis at the base of the superstructures supported by the foundation. The effect of the temperature load in the basemat is negligible and is not considered in the basemat analysis based on ACI 349. According to ACI 349, thermal gradients less than approximately 38 °C (100 °F) need not be analyzed because such gradients do not cause significant stress in the reinforcement or strength deterioration. In the NI common basemat, the temperature
gradient is approximately 50 °F and a uniform temperature change is less than 10 °C (50 °F).

The analysis of the foundation mat is performed by a three-dimensional finite element structure model, and the forces and moments determined in the analysis are input to the structural design.

The analysis and design of the foundations consider the effects of potential mat uplift, with particular emphasis on differential settlements of the basemat.

The foundation of the seismic Category I structure analysis is performed considering a soil/rock properties beneath the foundation as a nonlinear spring elements. The model is capable of determining the possibility of uplift of the basemat from the subgrade during postulated SSE events. The vertical spring at each node in the analytical model acts in compression only. The horizontal springs are active when the vertical spring is in compression and inactive when the vertical spring lifts off.

3.8.5.4.2 Analyses of Settlement during Construction

The basemat is analyzed and designed to consider settlements in various phases of construction.

The basemat is sufficiently reinforced to control stresses until the concrete placement of basemat walls and containment internal structure is completed.

3.8.5.4.2.1 Construction Sequence

The construction sequence analysis model consists of a foundation media considering 11 layers (soil layer model), NI common basemat (up to El. 78 ft for RCB and 55 ft for Auxiliary Building), and superstructure model (Auxiliary building, Internal structure, and Shell & Dome). The concrete used in the analysis is normal weight concrete with a compressive strength of 5,000 psi at 91 days and 6,000 psi at 91 days for the NI common basemat and superstructures. The concrete strength is assumed at the four hardening conditions to consider change of strength due to the concrete pouring sequence.

For the construction sequence within the NI common basemat, 19 basemat concrete segments are determined based on the concrete placement and hardening stages. For the construction sequence within the superstructure (Auxiliary Building, Containment Shell & Dome, Containment internal structure), the five segments of the superstructure are determined to consider the direction of construction sequence.
The construction sequence analysis, including NI building structure and superstructure, is performed for a total of 58 construction sequences. In addition, the construction sequence of the superstructure considers two cases to check the possibility of different cases (Case 1: Counterclockwise, Case 2: Clockwise). To consider the equipment load, the RCS weight is applied to each node when the slab, located at 156 ft, is hardening.

For post construction, the settlement due to creep in soil profile S01 is considered. Conversely, in soil profile S08, creep is not considered because the entire profile is rock. To consider the effect of creep in soil, the equation based on several experimental results suggested by Schmertmann (1970) is applied:

\[ C_2 = 1 + 0.2 \log_{10}(t/0.1) \] where, t is time, in years

To consider the effects from the construction sequence, the differences in member forces between the construction sequence analysis and the reference analysis are considered in the design.

A detailed description and results for the construction sequence are described in Technical Report, APR1400-E-S-NR-14006-P (Reference 40).

3.8.5.4.2.2 Various Settlement

For various types of settlement for the NI, the maximum vertical settlement, titling settlement, differential settlement between structures, and angular distortion are determined under construction and post-construction phases. The following three sequences are considered key for checking the four types settlements. For the EDGB and DFOT, the maximum vertical settlement and differential settlement between structures are determined. These settlements are used to define settlement criteria to be checked by the COL applicant to ensure that the design is adequate at the site.

Sequence No. 22: Completion of construction of NI common basemat.

Sequence No. 58 (Construction phase): Completion of construction of all superstructures.

Sequence No. 59 (Post-construction phase): End of plant life time, considering creep effect of soil.

a. Maximum vertical settlement
Maximum vertical displacement is the maximum calculated vertical deformation for the construction and post-construction phases under sequence No.58 and 59, respectively. Table 3.8-12 summarizes the maximum vertical settlement under construction and post-construction phases. For the maximum vertical settlement of the EDGB and DFOT buildings, it is determined from analysis that the construction sequence is not required. In order to consider characteristics of post-construction, the soil spring reflected in the equation described in section 3.8.5.4.2.1 is considered. Table 3.8-12 summarizes the maximum vertical settlement under construction and post-construction.

b. Maximum Tilting settlement

Tilting settlement is calculated as the ratio of the differential vertical settlement at the opposite edges of the buildings to the length between two edges. In the construction sequence, the maximum tilting settlement for the construction and post-construction phases are determined by the following equation under sequence No.58, end of construction, and No. 59, post-construction. Table 3.8-13 summarizes the maximum tilting settlement under construction and post-construction phases. For the EDGB and DFOT buildings, construction sequence and tilting settlement analysis are not needed because these buildings are relatively small and simple structures, and there are sufficient gaps between the EDGB/DFOT buildings and NI buildings.

$$\text{Maximum tilting settlement} = \text{arctan} (\Delta Uz/L)$$

c. Maximum differential settlement between structures

The maximum differential settlement between adjacent structures, it is determined based on the vertical displacement obtained from adjacent nodes of each structure. For the maximum differential settlement between structures under construction and post-construction, vertical settlement of the NI basemat is obtained from sequences No.58 and No.59. However, vertical settlement of the EDGB/DFOT is obtained from the analysis which does not consider construction sequence analysis since the other structures (i.e., EDGB/ DFOT) are not required. The construction sequence analysis is not needed because these buildings are relatively small and simple structures, and there are sufficient gaps between the EDGB/DFOT buildings and NI buildings. Table 3.8-14 summarizes the
maximum differential settlement between the structures under the construction and post-construction.

d. Maximum Angular distortion

Maximum angular distortion ($\beta = \delta / L$) is a measure of the differential vertical displacement between two adjacent points separated by the distance, $L$. To determine the angular distortion, three sequences (sequence No.22, No.58, and No.59) are selected for soil profile S01 and two sequences (sequence No.22 and No.58) for soil profile S08. Based on the deformation results from each sequence (Nos.22, 58, and 59), 12 groups are determined as check points for angular distortion. For checking angular distortion of each group corresponding to the soil profiles (S01, S08), angular distortion is plotted by vertical displacement along the distance between adjacent nodes within each group.

Detailed description and displacement graphs for angular distortion results are provided in Technical Report, APR1400-E-S-NR-14006-P (Reference 40). These vertical displacement graphs are used as the acceptance criteria for the COL applicant, not as maximum angular distortion values since these figures show the curvature of the entire basemat.

For the EDGB and DFOT buildings, construction sequence and maximum angular distortion are not needed because these buildings are relatively small and simple structures, and there are sufficient gaps between the EDGB/DFOT buildings and NI buildings.

3.8.5.4.3 Design Summary Report

A design summary report for the basemats is presented in Appendix 3.8A, where the design of representative critical sections of the structures is described. The detailed description of stability, uplift, settlement, bearing pressure check is presented in Appendix 3.8A. A detailed description of analysis for design and stability check is presented in Technical Report, APR1400-E-S-NR-14006-P (Reference 40).

The evaluation considering the deviations of as-procured or as-built construction to the design will be performed with the acceptance criteria, as described in Technical Report, APR1400-E-S-NR-14006-P (Reference 40).
3.8.5.5 Structural Acceptance Criteria

The structural acceptance criteria for the containment and other seismic Category I structures excluding the reactor containment building are described in Subsections 3.8.1.5 and 3.8.4.5, respectively. In particular, the acceptance criteria for the stability of seismic Category I structures are checked together with the structural acceptance criteria against the design loadings. The overturning, sliding, and flotation are checked as a minimum for stability of the basemat.

The design soil conditions are as provided in Section 2.5. The COL applicant is to provide reasonable assurance that the design criteria listed in Table 2.0-1 are met or exceeded (COL 3.8(12)).

The acceptance criteria for overturning, sliding, and flotation are described in Table 3.8-8. The factor of safety to design load combinations is calculated as stated below and compared to the minimum factors to provide reasonable assurance of the stability of the basemats.

3.8.5.5.1 Overturning Acceptance Criteria

The factor of safety against overturning is identified as the ratio of the resisting moment on overturning ($M_r$) to the overturning moment ($M_o$). Therefore, $FS_o = \frac{M_r}{M_o}$, not less than the factor of safety determined from Table 3.8-8.

Where:

- $FS_o =$ structure factor of safety against overturning caused by the design basis wind, tornado, or earthquake load
- $M_r =$ resisting moment determined as the dead load of the structure minus buoyant force from normal design groundwater table, multiplied by the distance from the structure edge to the structure center of gravity provided there is no overstress at the edge of the structure
- $M_o =$ overturning moment caused by wind, tornado, or earthquake

Resistance moment due to passive soil pressure is not included in $M_r$. Therefore, active and overburden soil pressures are also not considered.
3.8.5.5.2 Sliding Acceptance Criteria

The factor of safety against sliding caused by wind, tornado, or earthquake is identified by the following ratio:

\[ FS_s = \frac{F_s}{F_d}, \]

not less than the factor of safety determined from Table 3.8-8

Where:

- \( FS_s \) = structure factor of safety against sliding caused by wind, tornado, or earthquake
- \( F_s \) = sliding resistance along bottom of the basemat determined as the dead load of the structure minus the buoyant force from the normal design groundwater table and sliding resistance by shear key effect
- \( F_d \) = sliding force caused by wind, tornado, or earthquake load

The factor of safety against sliding is calculated by the ratio of resisting force to driving force. The factor of safety of NI common basemat against sliding caused by earthquake may be calculated at each time step for each soil case, i.e., by linear time history method. In this case, the minimum value is selected as the factor of safety.

The driving force is calculated from seismic horizontal force of the structure. From the time history analysis result, the total sum of seismic horizontal force of the structure is obtained for E-W and N-S direction, respectively. At each time step, resultant horizontal driving force is calculated from the E-W and N-S direction forces by square root of sum of their squares.

The resisting force consists of two categories: resisting force by base friction and resisting force by shear keys. The resisting force by base friction is based on the minimum friction force between the sliding interfaces. In the calculation of the resistant force, the coefficient of friction of 0.55 is used. It is based on that the coefficient of friction between waterproofing membrane and lean concrete is the minimum value among the interfaces of dissimilar materials.

The COL applicant is to verify that the coefficient of friction between the lean concrete and the supporting medium at the site is equal to or higher than 0.55. In order to meet this...
requirement, the COL applicant is to determine the specific undulation pattern in Figure 3.8-27 for two perpendicular horizontal directions (COL 3.8(13)). The minimum angle of internal friction of supporting medium is 35 degrees, which leads to a coefficient of friction of 0.7, and this is to be confirmed by the COL applicant (COL 2.5(10)). The coefficient of friction between the lean concrete and foundation concrete may be used as 1.0 or higher because construction joints of APR1400 shall be intentionally roughened.

The resisting force by base friction is calculated by multiplication of effective dead weight and coefficient of friction. For the calculation of the effective dead weight, probable adverse effects of the buoyant force from design ground water level and seismic uplift force are considered.

For the resisting force by shear keys, partial concave and convex areas of the basemat that are expected to play a role as shear keys are considered. Shear keys may be used to provide additional resistance against basemat sliding. In this sliding evaluation, the difference of passive soil pressure and active soil pressure are considered as the additional resistance provided the direct shear strength on the sliding soil face is larger than the force by passive soil.

3.8.5.5.3 Flotation Acceptance Criteria

The factor of safety against flotation is identified as the ratio of the total dead load of the structure including basemat ($D_r$) to the buoyant force ($F_b$). Therefore, $FS_f = D_r / F_b$, not less than the factor of safety determined from Table 3.8-8.

Where:

- $FS_f$ = structure factor of safety against flotation caused by the maximum design basis flood or groundwater table
- $D_r$ = total dead load of the structure including basemat
- $F_b$ = buoyant force caused by the design basis flood or high groundwater table, whichever is greater
Material, Quality Control, and Special Construction Techniques

The materials, quality control, special construction techniques, and tolerances for foundations conform with those set forth for the superstructures as discussed in Subsections 3.8.1.6 and 3.8.4.6 and Appendix 3.8A.

The COL applicant is to confirm that uneven settlement due to construction sequence of the NI basemat falls within the values specified in Table 2.0-1 (COL 3.8(14)).

Testing and Inservice Inspection Requirements

Testing and inservice surveillance of the basemat are performed in accordance with the requirements described in Subsections 3.8.1.7 and 3.8.4.7.

The COL applicant is to provide a site-specific monitoring program and to monitor maximum vertical settlement, differential settlement, tilt, and angular distortion to ensure they are less than the criteria in Table 2.0-1, Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d during construction and plant operation (COL 3.8(15)).

The COL applicant is to provide testing and inservice inspection programs to examine inaccessible areas of concrete structures for degradation and to monitor groundwater chemistry (COL 3.8(16)).

The long-term settlement is the site-specific characteristics. The COL applicant is to provide the soil parameters for APR1400 site (COL 3.8(17)).

SSCs between structures

Differential settlements (76.2 mm (3.0 in) shown in DCD Table 2.0-1) between structures (NI and EDG, NI and DFOT, EDG and DFOT) are used for design of SSCs under static loads.

The SSCs between buildings or structures are designed to allow the relative displacement or resist the forces due to relative movements of buildings. For this purpose, the floor response spectra and seismic anchor motions are used in the following procedure.

1) Seismic analyses of building structures are performed to compute floor response spectra and seismic anchor motions for the SSCs design.
2) SSCs shall have the support details to allow relative movements of between buildings or structures.

3) The seismic anchor motions between buildings or structures are considered in the SSCs design.

3.8.6 Combined License Information

COL 3.8(1) The COL applicant is to perform concrete long-term material testing in a way which verifies physical properties of materials used during the design stage and the characteristics of long term deformation of concrete.

COL 3.8(2) The COL applicant is to provide the detailed design results and evaluation of the ultimate pressure capacity of penetrations, including the equipment hatch, personnel airlocks, electrical and piping penetrations in accordance with NRC RG 1.216.

COL 3.8(3) The COL applicant is to provide detailed analysis and design procedure for the equipment hatch, personnel airlocks, and electrical penetrations.

COL 3.8(4) The COL applicant is to provide detailed analysis and design procedure for the transfer tube penetration assembly.

COL 3.8(5) The COL applicant is to provide the design of site-specific seismic Category I structures such as the essential service water building and the component cooling water heat exchanger building, essential service water conduits, component cooling water piping tunnel, and class 1E electrical duct runs.

COL 3.8(6) The COL applicant is to evaluate any applicable site-specific loads such as explosive hazards in proximity to the site, projectiles and missiles generated from activities of nearby military installations, potential non-terrorism related aircraft crashes, and the effects of seiches, surges, waves, and tsunamis.

COL 3.8(7) The COL applicant is to perform the analysis and design of the steel plate for the new fuel storage pit.
COL 3.8(8) The COL applicant is to determine the environmental condition associated with the durability of concrete structures and provide the concrete mix design that prevents concrete degradation including the reactions of sulfate and other chemicals, corrosion of reinforcing bars, and influence of reactive aggregates.

COL 3.8(9) The COL applicant is to determine construction techniques to minimize the effects of thermal expansion and contraction due to hydration heat, which could result in cracking.

COL 3.8(10) For safety and serviceability of seismic Category I structures during the operation of the plant, the COL applicant is to provide appropriate testing and in-service inspection programs to examine the condition of normally inaccessible, below-grade concrete for signs of degradation and to conduct periodic site monitoring of ground water chemistry. In-service inspection of the accessible portion of concrete structures is also to be performed.

COL 3.8(11) The COL applicant is to verify that the coefficient of friction between the lean concrete and waterproofing membrane is greater than or equal to 0.55.

COL 3.8(12) The COL applicant is to provide reasonable assurance that the design criteria listed in Table 2.0-1 are met or exceeded.

COL 3.8(13) The COL applicant is to verify that the coefficient of friction between the lean concrete and the supporting medium at the site is equal to or higher than 0.55. In order to meet this requirement, the COL applicant is to determine the specific undulation pattern in Figure 3.8-27 for two perpendicular horizontal directions.

COL 3.8(14) The COL applicant is to confirm that uneven settlement due to construction sequence of the NI basemat falls within the values specified in Table 2.0-1.

COL 3.8(15) The COL applicant is to provide a site-specific monitoring program and to monitor maximum vertical settlement, differential settlement, tilt, and angular distortion to ensure they are less than the criteria in Table 2.0-1, Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d during construction and plant operation.
The COL applicant is to provide testing and inservice inspection programs to examine inaccessible areas of concrete structures for degradation and to monitor groundwater chemistry.

The COL applicant is to provide the following soil information for the APR1400 site: 1) elastic shear modulus and Poisson's ratio of the subsurface soil layers, 2) consolidation properties including data from one-dimensional consolidation tests (initial void ratio, Cc, Ccr, OCR, and complete e-log p curves) and time-versus-consolidation plots, 3) moisture content, Atterberg limits, grain size analyses, and soil classification, 4) construction sequence and loading history, and 5) excavation and dewatering programs.

A detailed construction sequence analysis to determine the resulting construction settlements, including the various standard soils profiles (S01-S04, S06-S09) and sequencing of concrete pours for the NI common basemat (RCB and Auxiliary Building), and superstructure model (Auxiliary Building, internal structures, and Shell & Dome), is presented in Section 3.8.5.4.2. A comparison of the four types of construction settlements (i.e. maximum vertical settlement, tilting settlement, maximum differential settlement between structures, and angular distortion) to the maximum criteria is summarized in Tables 3.8-12 through 3.8-14, and section 3.8.5.4.2.2.d.

The COL applicant should use the construction sequence settlement analysis given in Section 3.8.5.4.2, substituting site-specific soil layer conditions, to ensure that the four types of settlement criteria described in Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d are satisfied. An alternative construction sequence and settlement analysis may be performed by the COL applicant in response to 1) the inability to meet the settlement criteria described in Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d using the DCD approach discussed in Section 3.8.5.4.2 or 2) Other site specific factors that may require a different construction plan and foundation sequence. However, in either case the COL applicant shall satisfy four types of settlement criteria described in Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d.
COL 3.8(19) The following items should be considered by the COL applicant:

1) The surveyed soil profiles will be developed.

2) Based on the surveyed soil characteristics, differences from the DCD soil profiles may exist.

These differences may include:

a. Stiff or soft soil areas;

b. Different soil types (e.g., cohesive);

c. Potential for loss of cement in the mudmat;

d. Non-uniformity of soil layers, or

e. Other differences in the soil profile from the properties assumed in design certification.

If any of these items and/or conditions are identified, then a site-specific evaluation\(^1\) shall be performed and checked for adequacy.

3) The time (i.e., short term and long term) instantaneous settlement and time-consolidation effects shall be evaluated in accordance with surveyed soil profiles regardless if a site-specific evaluation is needed under Item 2) above. The bearing pressure shall be checked to demonstrate acceptability with the acceptance criteria in DCD Table 2.0-1. Settlements shall be checked in Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d.

4) The COL applicant will build the seismic Category I structure according to the construction sequence used in the site-specific construction sequence analysis.

\(^1\)evaluation includes basemat and superstructure design (forces/stresses), settlement evaluations, soil bearing pressure evaluation, and stability evaluation.
5) If a site-specific evaluation1) is required, the COL applicant should perform a construction sequence analysis based on the site-specific parameters. If the settlement including results of construction sequence analysis exceeds the acceptance criteria described in Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d the construction sequence will be modified to meet the acceptance criteria described in Table 3.8-12 thru Table 3.8-14, and section 3.8.5.4.2.2.d by COL applicant.

6) The effect on the design of seismic Category I structures due to construction sequence analysis shall be accounted for by the COL applicant.

COL 3.8(20) The COL applicant shall perform site-specific evaluations if the shear wave velocity is less than the shear wave velocity profile used in the various basemat evaluations for design certification. The site-specific evaluations ([settlement (maximum vertical displacement, tilt, differential settlement between structures, angular distortion), soil bearing pressure (static and dynamic loading cases), overturning, and sliding]) and 3D FEM global analysis for basemat design of seismic Category I structures shall be performed using the site-specific parameters (measured $E_{\text{static}}$, $E_{\text{dynamic}}$ consistent with soil strain assumed in SSI analysis) and the methodology described in DCD Tier 2, Subsection 3.8.5 and Technical Report, APR1400-E-S-NR-14006-P, Subsection 4.

COL 3.8(21) The COL applicant is to confirm that the parking position of the crane and trolley when the crane is not being used is: location of polar crane: Az.280°, trolley location: 12ft 7in away from end of east part. The COL applicant is to confirm that this requirement is included in the technical specification of the COL application for the use of the polar crane.

COL 3.8(22) The COL applicant may provide construction tolerance acceptance criteria and the basis for the criteria (e.g., through the use of analysis, industry research, or testing) for cases where the tolerances in ACI 117 and ANSI/AISC 303, for structural concrete and structural steel, respectively, may be exceeded.
3.8.7 References


41. ACI 349, “Code Requirements for Nuclear Safety-Related Concrete Structures,” American Concrete Institute, 1997.


53. ACI 318, “Building code requirements for reinforced concrete,” American Concrete Institute, 2008.

54. ACI 301, “Specifications for Structural Concrete for Building,” American Concrete Institute, 2010.


57. AISI S100, “Specification for the Design of Cold-Formed Steel Structural Members,” American Iron and Steel Institute, 2012.

58. ACI 211.1, “Standard Practice for Selecting Proportions for Normal, Heavy Weight, and Mass Concrete,” American Concrete Institute, 1991 (R2007).

59. ACI 214, “Recommended Practice for Evaluation of Strength Test Results of Concrete,” American Concrete Institute, 1991 (R1997).

60. ACI 304 R, “Guide for Measuring, Mixing, Transporting, and Placing Concrete,” American Concrete Institute, 2000 (R2009).

61. ACI 305 R, “Hot Weather Concreting,” American Concrete Institute, 1999.

63. ACI 308, “Standard Practice for Curing Concrete,” American Concrete Institute, 1992 (R1997).

64. ACI 309 R, “Guide for Consolidation of Concrete,” American Concrete Institute, 2005.

65. ACI 311.1 R, “ACI Manual of Concrete Inspection,” American Concrete Institute, 1999.

66. ACI 315, “Details and Detailing of Concrete Reinforcement,” American Concrete Institute, 1999.


77. ACI 340R, “Design of Structural Reinforced Concrete Elements in accordance with the Strength Design Method of ACI 318-95,” American Concrete Institute, 1997.

78. ACI 117, “Specification for Tolerances for Concrete Construction and Materials and Commentary,” American Concrete Institute, 2010.

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#### Codes, Standards, Specifications, and Regulations

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Abbreviation

ACI American Concrete Institute
AISC American Institute of Steel Construction
AISI American Iron and Steel Institute
ANS American Nuclear Society
ASME American Society of Mechanical Engineers
ASTM American Society of Testing and Materials
AWS American Welding Society
# Seismic Category I Structure Load Combination for the Reactor Containment Building

| Category / Loading Condition | No | D<sup>(1)</sup> | L<sup>(2)</sup> | F | P<sub>i</sub> | G | T<sub>i</sub> | T<sub>o</sub> | E<sub>i</sub> | W | W<sub>t</sub> | R<sub>a</sub> | R<sub>s</sub> | Y<sub>f</sub> | Y<sub>i</sub> | Y<sub>m</sub> | Y<sub>t</sub> | H | H<sub>a</sub> | P<sub>y</sub> | H<sub>s</sub> | P<sub>s</sub> |
|-----------------------------|----|-------------|-------------|--|-------|---|-------|-------|-------|---|-------|-------|-------|-------|-------|-------|-------|---|-------|-------|-------|
| Serv                        |    |             |             |  |       |   |       |       |       |   |       |       |       |       |       |       |       |   |       |       |       |
| Test                        | 1  | 1.0         | 1.0         | 1.0 | 1.0   | - | -     |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
| Construction                | 2  | 1.0         | 1.0         | 1.0 | -     | - | -     | 1.0   | -     | - | 1.0   | -     | -     |       |       |       |       | - | -     |       |       |
| Normal                      | 3  | 1.0         | 1.0         | 1.0 | -     | 1.0 | -     |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
| Severe environmental        | 4  | 1.0         | 1.3         | 1.0 | -     | 1.0 | -     |       |       | - | 1.0   |       | -     |       |       |       |       | - | -     |       |       |
|                            | 5  | 1.0         | 1.3         | 1.0 | -     | 1.0 | -     |       |       | - | -     |       | -     |       |       |       |       | - | 1.5   |       |       |
| Extreme environmental       | 6  | 1.0         | 1.0         | 1.0 | -     | 1.0 | -     |       |       | - | 1.0   |       | -     |       |       |       |       | - | -     |       |       |
|                            | 7  | 1.0         | 1.0         | 1.0 | -     | 1.0 | -     |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
|                            | 8  | 1.0         | 1.0         | 1.0 | -     | 1.0 | -     |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
| Abnormal                    | 9  | 1.0         | 1.0         | 1.0 | -     | 1.0 | 1.5   |       |       | - | 1.0   |       | -     |       |       |       |       | - | -     |       |       |
|                            | 10 | 1.0         | 1.0         | 1.0 | -     | 1.0 | 1.0   |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
|                            | 11 | 1.0         | 1.0         | 1.0 | -     | 1.0 | 1.0   |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
|                            | 12 | 1.0         | 1.0         | 1.0 | -     | 1.0 | 1.25  |       |       | - | 1.0   |       | -     |       |       |       |       | - | 1.25  |       |       |
| Abnormal/severe environmental | 13 | 1.0     | 1.0         | 1.0 | -     | 1.0 | 1.0   |       |       | - | 1.0   |       | -     |       |       |       |       | - | -     |       |       |
| Abnormal/extreme environmental | 14 | 1.0      | 1.0         | 1.0 | -     | 1.0 | -     |       |       | - | -     |       | -     |       |       |       |       | - | -     |       |       |
| Combustible Gas Control inside Containment<sup>(3)</sup> | 15 | 1.0      | -           | 1.0 | -     | -   | -     |       |       | - | -     |       | -     |       |       |       |       | - | -     |       | 1.0   |

(1) D<sub>i</sub> in Table 3.8-8 is included in D.

(2) Includes all temporary construction loads during and after construction of the containment; also includes L<sub>v</sub> in Table 3.8-8 and C.

(3) The strain does not exceed the values given in ASME Section III, Division 2, Table CC-3720-1.
### Load Definitions and Load Combinations for ASME Class MC Components

| No. | Load Condition | D | L | T | R | P | T | R | P | T | R | P | E | T | R | P | T | Rs | Ps | Ts | Yr | Yj | Ym | FL | Stress Intensity Limits
| 1   | Testing Condition | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| 2   | Design Condition | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| 3   | Service Condition | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |

1. Normal operating plant condition
2. Operating plant condition in conjunction with the actuation of multiple SRVs
3. Design-basis LOCA
4. Multiple SRV actuations in conjunction with small or intermediate break accident
5. Operating plant condition in combination with OBE
6. Design-basis LOCA in combination with OBE and multiple SRV actuations
7. Operating plant condition in combination with SSE
8. Multi-SRV actuations in combination with small- or intermediate-break accident and SSE
9. Dead load plus pressure resulting from an accident that releases hydrogen
10. Design-basis LOCA in combination with SSE and local dynamic loadings
11. Multiple SRV actuations in combination with small- or intermediate-break accident, SSE, and local dynamic loading
12. Dead load plus pressure resulting from an accident that releases hydrogen

(1) The loads may be combined by their actual time history of occurrence taking into consideration their dynamic effect upon the structure.
(2) Acceptance criteria are taken from the referenced section in Section III of the ASME Code
(3) Definitions of the terms in this table are as follows:
- **D**: Dead loads
- **L**: Live loads, including all loads resulting from platform flexibility and deformation and from crane loading, if applicable
- **P**: Test pressure
- **T**: Test temperature
- **R**: Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition
- **Rt**: Pipe reactions during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition
- **E**: External pressure loads resulting from pressure variation either inside or outside containment
- **Eo**: Loads generated by the OBE, including sloshing effects, if applicable
- **Eh**: Loads generated by the SSE, including sloshing effects, if applicable
- **Ej**: Pressure load generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures), pool swell, and subsequent hydrodynamic loads
- **Eo**: Thermal loads generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads
- **Es**: Pipe reactions under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads
- **Ep**: Pressure loads generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures), pool swell, and subsequent hydrodynamic loads
- **Eo**: All pressure loads that are caused by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic loads, if applicable
- **Eh**: All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads, if applicable
- **Er**: All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic reaction loads, if applicable
- **Y**: Equivalent static load on the structure generated by the reaction on the broken pipe during the design-basis accident
- **Yr**: Equivalent static load on the structure generated by the reaction on the broken pipe during the design-basis accident
- **Ym**: Equivalent static load on the structure generated by or during the design-basis accident, such as pipe whipping
- **Ft**: Load generated by the post-LOCA flooding of the containment, if applicable
- **Pp**: Pressure load generated from 100-percent fuel cladding water reaction

### Footnotes
- (1) The loads may be combined by their actual time history of occurrence taking into consideration their dynamic effect upon the structure.
- (2) Acceptance criteria are taken from the referenced section in Section III of the ASME Code
- (3) Definitions of the terms in this table are as follows;
Table 3.8-4

Physical Properties for Materials To Be Used for Pressure Parts
or Attachment to Pressure Part ASME Code Class MC Components

<table>
<thead>
<tr>
<th>Material Specification</th>
<th>$S_u$ Minimum Ultimate Tensile MPa (ksi)</th>
<th>$S_y$ Minimum Yield at Ambience MPa (ksi)</th>
<th>$S_y$ Minimum Yield at 171 °C (340 °F) MPa (ksi)</th>
<th>$S_m$ ASME Code Allowable Stress Intensity at 171 °C (340 °F) MPa (ksi)</th>
<th>Note</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plate</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SA-516 Gr 70</td>
<td>483 (70)</td>
<td>262 (38)</td>
<td>229 (33.26)</td>
<td>153 (22.18)</td>
<td></td>
</tr>
<tr>
<td>SA-516 Gr 60</td>
<td>414 (60)</td>
<td>221 (32)</td>
<td>193 (27.94)</td>
<td>129 (18.66)</td>
<td></td>
</tr>
<tr>
<td>SA-240 Type 304</td>
<td>517 (75)</td>
<td>207 (30)</td>
<td>150 (21.78)</td>
<td>134 (19.48)</td>
<td></td>
</tr>
<tr>
<td>Pipe</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SA-106 Gr B</td>
<td>414 (60)</td>
<td>241 (35)</td>
<td>211 (30.6)</td>
<td>138 (20.0)</td>
<td></td>
</tr>
<tr>
<td>SA-333 Gr 6</td>
<td>414 (60)</td>
<td>241 (35)</td>
<td>211 (30.6)</td>
<td>138 (20.0)</td>
<td></td>
</tr>
<tr>
<td>SA-312 Type 304</td>
<td>517 (75)</td>
<td>207 (30)</td>
<td>150 (21.78)</td>
<td>134 (19.48)</td>
<td></td>
</tr>
<tr>
<td>Forgings and fittings</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SA-350 LF-2</td>
<td>483 (70)</td>
<td>248 (36)</td>
<td>217 (31.5)</td>
<td>144 (20.94)</td>
<td></td>
</tr>
<tr>
<td>SA-182 F22 Class 3</td>
<td>517 (75)</td>
<td>310 (45)</td>
<td>268 (38.84)</td>
<td>168 (24.34)</td>
<td></td>
</tr>
<tr>
<td>Bolting</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SA-193 B7</td>
<td>793 (115)</td>
<td>655 (95)</td>
<td>579 (83.98)</td>
<td>193 (28.0)</td>
<td>Between 6.35 and 10.16 cm (2.5 and 4 in.) diameter</td>
</tr>
<tr>
<td>SA-320 Gr L43</td>
<td>862 (125)</td>
<td>724 (105)</td>
<td>642 (93.06)</td>
<td>214 (31.04)</td>
<td>Under 6.35 cm (2.5 in.) diameter</td>
</tr>
</tbody>
</table>
Table 3.8-5 (1 of 2)
Design Loads for Seismic Category I Structures

<table>
<thead>
<tr>
<th>Structures</th>
<th>Loadings kN/m² (psf)</th>
<th>Remarks</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Dead Load (D)</td>
<td>Live Load (L)</td>
<td>Rain and Wind (L and W)</td>
</tr>
<tr>
<td>Containment Internal Structure</td>
<td>*</td>
<td>*</td>
<td>N/A</td>
</tr>
<tr>
<td>Primary Shield Wall</td>
<td>-</td>
<td>-</td>
<td>N/A</td>
</tr>
<tr>
<td>Operating Floor, El. 156'-0&quot;</td>
<td>-</td>
<td>50.0 (1,000)</td>
<td>N/A</td>
</tr>
<tr>
<td>Intermediate Floor, El. 136'-6&quot;</td>
<td>-</td>
<td>10.0 (200)</td>
<td>N/A</td>
</tr>
<tr>
<td>Intermediate Floor, El. 114'-0&quot;</td>
<td>-</td>
<td>10.0 (200)</td>
<td>N/A</td>
</tr>
<tr>
<td>Secondary Shield Wall</td>
<td>-</td>
<td>-</td>
<td>N/A</td>
</tr>
</tbody>
</table>
### Table 3.8-5 (2 of 2)

<table>
<thead>
<tr>
<th>Structures</th>
<th>Loadings kN/m² (psf)</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Dead Load (D)</td>
<td>Live Load (L)</td>
</tr>
<tr>
<td>Other Seismic Category I Structure except RCB</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>Interior walls</td>
<td>1.0 (20)</td>
<td>16.8 (350)</td>
</tr>
<tr>
<td>Exterior walls</td>
<td>0.5 (10)</td>
<td>16.8 (350)</td>
</tr>
<tr>
<td>Roof slabs</td>
<td>-</td>
<td>4.8 (100)</td>
</tr>
<tr>
<td>Main floors</td>
<td>-</td>
<td>10.0 (200) ~ 24.0 (500)</td>
</tr>
<tr>
<td>Basemat at elevation 55 ft 0 in</td>
<td>-</td>
<td>24.0 (500)</td>
</tr>
<tr>
<td>Cask loading and decontamination pit</td>
<td>-</td>
<td>50.0 (1,000)</td>
</tr>
</tbody>
</table>

1. The masses of all structures are included in all load combinations as dead loads.
2. All structures are designed for seismic loads.
3. See Subsection 3.8.4.3 for design soil loads, including groundwater, thermal loads, wind loads, tornado or hurricane loads, and added live load due to precipitation.
4. Abnormal loads due to main steam and feedwater line breaks are considered.
5. Loads for SG removal (1,000 psf) are considered at elevation 156 ft 0 in CVCS area.
6. Extreme external temperatures are evaluated to determine temperatures to be combined with extreme internal temperatures.
7. Soil surcharge load on exterior walls due to construction loads.
8. Live load on shear wall in horizontal (out-of-plane) direction to account for attachment loads.
9. Snow drifts are considered for live load on lower roofs.
10. Normal live load is 200 psf and equipment removal load is 500 psf.
11. 50.0 kN/m² (1,000 psf) is for movable equipment during construction and maintenance.
## Table 3.8-6

Types and Applicable Loading Conditions for Dead Loads and Live Loads

<table>
<thead>
<tr>
<th>Applicable Loading Conditions</th>
<th>Load</th>
<th>Definition of Loads</th>
</tr>
</thead>
<tbody>
<tr>
<td>Construction Test Normal Severe Environmental Abnormal Extreme Environmental Abnormal/Severe Environmental Abnormal/Extreme Environmental</td>
<td></td>
<td></td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>D&lt;sub&gt;h&lt;/sub&gt; – Vertical Pressure of liquids (with due regard to variations in the liquid depth)</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>D&lt;sub&gt;a&lt;/sub&gt; – Self-weight of structure including waterproofing, siding, and insulation</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>D&lt;sub&gt;e&lt;/sub&gt; – Weight of equipment and its contents (gravity load under operating conditions). This includes crane self-weights and trolley hoist self-weights.</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>D&lt;sub&gt;m&lt;/sub&gt; – Shoring and other loads provided by contractor</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>L&lt;sub&gt;h&lt;/sub&gt; – Hydrostatic loads due to weight and pressures of fluids with well-defined densities and controllable maximum height</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>L&lt;sub&gt;g&lt;/sub&gt; – Loads due to the weight and pressure of soil, water in soil, or other materials</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>L&lt;sub&gt;s&lt;/sub&gt; – Snow loads</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>L&lt;sub&gt;f&lt;/sub&gt; – Floor and roof live loads</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>L&lt;sub&gt;o&lt;/sub&gt; – Operating reaction of equipment excluding D&lt;sub&gt;e&lt;/sub&gt;</td>
</tr>
<tr>
<td>×</td>
<td></td>
<td>L&lt;sub&gt;e&lt;/sub&gt; – 2.4 kN/m&lt;sup&gt;2&lt;/sup&gt; (50 psf) seismic live load for Floor and roof during SSE loadings</td>
</tr>
</tbody>
</table>

Rev. 3
Table 3.8-7A

Seismic Category I Structures Excluding Containment Structure

Reinforced Concrete – Ultimate Strength Design Load Combination Table

<table>
<thead>
<tr>
<th>Loading Condition</th>
<th>No</th>
<th>Normal</th>
<th>Severe Environmental</th>
<th>Abnormal</th>
<th>POSRV Load</th>
<th>Extreme Environmental</th>
<th>Design Strength</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>D(1)</td>
<td>Dd</td>
<td>L(3)</td>
<td>Lk</td>
<td>Tg</td>
<td>Rs</td>
</tr>
<tr>
<td>Construction</td>
<td>1</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>-</td>
<td>0.9</td>
<td>-</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
</tr>
<tr>
<td>Test</td>
<td>3</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
<td>1.1</td>
<td>1.3</td>
<td>1.3</td>
</tr>
<tr>
<td>Normal</td>
<td>4</td>
<td>1.4</td>
<td>-</td>
<td>1.7</td>
<td>1.4</td>
<td>-</td>
<td>1.7</td>
</tr>
<tr>
<td></td>
<td>5</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
<td>1.1</td>
<td>1.2</td>
<td>1.3</td>
</tr>
<tr>
<td>Severe Environmental</td>
<td>6</td>
<td>1.4</td>
<td>-</td>
<td>1.7</td>
<td>1.4</td>
<td>-</td>
<td>1.7</td>
</tr>
<tr>
<td></td>
<td>7</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
<td>1.1</td>
<td>1.2</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>1.4</td>
<td>-</td>
<td>1.7</td>
<td>1.4</td>
<td>-</td>
<td>1.7</td>
</tr>
<tr>
<td></td>
<td>9</td>
<td>1.1</td>
<td>-</td>
<td>1.3</td>
<td>1.1</td>
<td>1.2</td>
<td>1.3</td>
</tr>
<tr>
<td>Abnormal</td>
<td>10</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>11</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Extreme Environmental</td>
<td>12</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>13</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>14</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Abnormal/Extreme Environmental</td>
<td>15</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>-</td>
</tr>
</tbody>
</table>

(1) Where a load occurs simultaneously with the other loads or is always present and reduces effects of other loads, the load factor is taken as 0.9; otherwise, the load factor is taken as zero.

(2) Hydrodynamic loads associated with seismic loads are included in Es.

(3) L includes Lg, Ls, Lf, L0, and Le loads.

(4) POSRV load (Pac) is used for IRWST design (Refer to SRP 3.8.1).

(5) 1.5 Pn should be used in place of 1.4 Pn for IRWST design.
## Seismic Category I Structures Structural Steel – Elastic Design Load Combination Table

<table>
<thead>
<tr>
<th>Loading Condition</th>
<th>No</th>
<th>Normal</th>
<th>Severe Environmental</th>
<th>Abnormal</th>
<th>Extreme Environmental</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>D</td>
<td>Dₖ</td>
<td>L(12)</td>
<td>Tₒ</td>
<td>Rₙ</td>
</tr>
<tr>
<td>Construction</td>
<td>1</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
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</tr>
<tr>
<td></td>
<td>3</td>
<td>-</td>
<td>0.75</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Test</td>
<td>4</td>
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<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Normal</td>
<td>5</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
</tr>
<tr>
<td>Severe Environmental(10),(11)</td>
<td>7</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
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</tr>
<tr>
<td></td>
<td>8</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>9</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Abnormal(4),(7)</td>
<td>10</td>
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<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>11</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>12</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
</tr>
<tr>
<td>Extreme Environmental</td>
<td>13</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>14</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>15</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Abnormal/Extreme Environmental(7),(8)</td>
<td>16</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
</tr>
</tbody>
</table>
Table 3.8-7B (2 of 2)

(1) All load combinations are checked for a no-live-load condition.
(2) For primary plus secondary stress, the allowable stresses are increased by a factor of 1.5.
(3) In load combinations 10 through 16, the design stress in shear is not to exceed \(1.4 \times \text{AISC N690}\) in members and bolts.
(4) The load combination 12 is to be used when the global (non-transient) sustained effects of \(T_s\) are considered.
(5) The design stress where axial compression exceeds 20 percent of normal allowable is \(1.5 \times \text{AISC N690}\) for load combinations 10, 11, 12, 13, 14, and 15 and 1.6 for load combination 16.
(6) In no instance does the allowable stress exceed 0.7 \(F_u\) in axial tension or 0.7 \(F_u\) times the ratio \(Z/S\) for tension plus bending.
(7) The maximum values of \(P_a, T_s, R_s, Y_j, Y_f\), and \(Y_{m}\), including an appropriate dynamic load factor, is used in load combination 11, 12, and 16, unless an appropriate time-history analysis is performed to justify otherwise.
(8) In combining loads from a postulated high-energy pipe break accident and a seismic event the SRSS (square root of the sum of the squares) may be used, provided the responses are calculated on a linear basis.
(9) Secondary stresses that are used to limit primary stresses are treated as primary stresses.
(10) Consideration shall also be given to snow and other loads as defined in ASCE 7.
(11) Allowable stress limits coefficients are applied to the basic stress allowable of AISC. The coefficients for AISC N690 are supplemented by the requirements identified in subsection 3.8.4.5.
(12) \(L\) includes \(L_b, L_g, L_s, L_f, L_0\) and \(L_e\) loads.
### Table 3.8-7C

**Fuel Storage Rack – Design Loading Combination Table**

<table>
<thead>
<tr>
<th>Load Combination</th>
<th>Acceptance Limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>$D + L$</td>
<td>ASME Section III, NF Level A Service Limits for Class 3</td>
</tr>
<tr>
<td>$D + L + T_o$</td>
<td>ASME Section III, NF Level B Service Limits for Class 3</td>
</tr>
<tr>
<td>$D + L + T_o + P_f$</td>
<td>ASME Section III, NF Level D Service Limits for Class 3</td>
</tr>
<tr>
<td>$D + L + T_a + E'$</td>
<td>The functional capability of the fuel racks should be demonstrated.</td>
</tr>
</tbody>
</table>

Where:
- $D$ : Deadweight including fuel assembly weight
- $L$ : Live load
- $E'$ : Safe shutdown earthquake (SSE)
- $T_o$ : Differential temperature-induced loads, based on the most critical transient or steady-state condition under normal operation or shutdown conditions
- $T_a$ : Highest temperature associated with the postulated abnormal design conditions
- $F_d$ : Force caused by the accidental drop of the heaviest load from maximum possible height
- $P_f$ : Upward force on the racks caused by a postulated stuck fuel assembly

Note:
For the APR1400, the operating basis earthquake (OBE) ground motion is defined as one-third the SSE ground motion design response spectra. Therefore, in accordance with 10 CFR Part 50, Appendix S, an OBE design analysis is not required and load combinations involving $E$ have been removed.
Table 3.8-8

Acceptance Criteria for Overturning, Sliding, and Flotation

<table>
<thead>
<tr>
<th>Load Combination</th>
<th>Minimum Factor of Safety</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Overturning</td>
</tr>
<tr>
<td>D+He+W</td>
<td>1.5</td>
</tr>
<tr>
<td>D+He+E_s</td>
<td>1.1</td>
</tr>
<tr>
<td>D+He+W_t</td>
<td>1.1</td>
</tr>
<tr>
<td>D+H_s</td>
<td>-</td>
</tr>
</tbody>
</table>

Where:
- **D**: Dead load
- **H_e**: Static and dynamic lateral and vertical earth pressure including buoyant effect of normal design groundwater table level
- **H_s**: Buoyant force of the design basis flood
- **W**: Wind load
- **W_t**: Tornado or hurricane load
- **E_s**: Seismic load
### Table 3.8-9 (1 of 2)

**List of Containment Penetrations Greater than 18 Inches**

<table>
<thead>
<tr>
<th>Elevation</th>
<th>Azimuth Deg-Min</th>
<th>Sleeve Data</th>
<th>Projection</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Length</td>
<td>Outside Dia.</td>
</tr>
<tr>
<td>142'00-0/0&quot;</td>
<td>044-00</td>
<td>03'11-3/4&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>147'00-0/0&quot;</td>
<td>055-00</td>
<td>03'11-3/4&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>146'06-0/0&quot;</td>
<td>060-00</td>
<td>03'11-1/4&quot;</td>
<td>24.000&quot;</td>
</tr>
<tr>
<td>146'06-0/0&quot;</td>
<td>063-30</td>
<td>03'11-1/4&quot;</td>
<td>24.000&quot;</td>
</tr>
<tr>
<td>142'00-0/0&quot;</td>
<td>063-30</td>
<td>03'11-3/4&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>146'06-0/0&quot;</td>
<td>116-30</td>
<td>03'11-1/4&quot;</td>
<td>24.000&quot;</td>
</tr>
<tr>
<td>142'00-0/0&quot;</td>
<td>116-30</td>
<td>03'11-3/4&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>146'06-0/0&quot;</td>
<td>120-00</td>
<td>03'11-1/4&quot;</td>
<td>24.000&quot;</td>
</tr>
<tr>
<td>147'00-0/0&quot;</td>
<td>125-00</td>
<td>03'11-3/4&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>142'00-0/0&quot;</td>
<td>136-00</td>
<td>03'11-3/4&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>121'00-0/0&quot;</td>
<td>262-39-32</td>
<td>09'01-1/8&quot;</td>
<td>53.750&quot;</td>
</tr>
<tr>
<td>108'00-0/0&quot;</td>
<td>311-30</td>
<td>05'00-0/0&quot;</td>
<td>20.000&quot;</td>
</tr>
<tr>
<td>109'00-0/0&quot;</td>
<td>008-00</td>
<td>05'00-0/0&quot;</td>
<td>20.000&quot;</td>
</tr>
<tr>
<td>116'00-0/0&quot;</td>
<td>311-30</td>
<td>05'00-0/0&quot;</td>
<td>24.000&quot;</td>
</tr>
<tr>
<td>109'00-0/0&quot;</td>
<td>307-30</td>
<td>05'00-0/0&quot;</td>
<td>28.000&quot;</td>
</tr>
<tr>
<td>129'00-0/0&quot;</td>
<td>348-00</td>
<td>05'00-0/0&quot;</td>
<td>20.000&quot;</td>
</tr>
<tr>
<td>129'00-0/0&quot;</td>
<td>352-00</td>
<td>05'00-0/0&quot;</td>
<td>20.000&quot;</td>
</tr>
<tr>
<td>122'06-0/0&quot;</td>
<td>004-00</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>122'06-0/0&quot;</td>
<td>008-00</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>129'00-0/0&quot;</td>
<td>275-30</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>129'00-0/0&quot;</td>
<td>279-00</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
</tr>
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<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>109'00-0/0&quot;</td>
<td>300-00</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
</tr>
<tr>
<td>115'06-0/0&quot;</td>
<td>228-30</td>
<td>05'00-0/0&quot;</td>
<td>26.000&quot;</td>
</tr>
</tbody>
</table>
### Table 3.8-9 (2 of 2)

<table>
<thead>
<tr>
<th>Elevation</th>
<th>Azimuth Deg-Min</th>
<th>Length</th>
<th>Outside Dia.</th>
<th>Thickness</th>
<th>Material</th>
<th>Projection</th>
</tr>
</thead>
<tbody>
<tr>
<td>109'00-0/0&quot;</td>
<td>232-30</td>
<td>05'00-0/0&quot;</td>
<td>28.000&quot;</td>
<td>1.250&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 10.50</td>
</tr>
<tr>
<td>109'00-0/0&quot;</td>
<td>227-30</td>
<td>05'00-0/0&quot;</td>
<td>20.000&quot;</td>
<td>0.500&quot;</td>
<td>SA-333 Gr-6</td>
<td>01.50 10.50</td>
</tr>
<tr>
<td>129'00-0/0&quot;</td>
<td>192-00</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
<td>0.375&quot;</td>
<td>SA-333 Gr-6</td>
<td>01.50 10.50</td>
</tr>
<tr>
<td>129'00-0/0&quot;</td>
<td>236-00</td>
<td>05'00-0/0&quot;</td>
<td>18.000&quot;</td>
<td>0.375&quot;</td>
<td>SA-333 Gr-6</td>
<td>01.50 10.50</td>
</tr>
<tr>
<td>190'06-0/0&quot;</td>
<td>229-00</td>
<td>05'00-0/0&quot;</td>
<td>60.000&quot;</td>
<td>1.000&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 10.50</td>
</tr>
<tr>
<td>190'06-0/0&quot;</td>
<td>239-00</td>
<td>05'00-0/0&quot;</td>
<td>60.000&quot;</td>
<td>1.000&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 10.50</td>
</tr>
<tr>
<td>147'06-0/0&quot;</td>
<td>011-08</td>
<td>08'08-1/4&quot;</td>
<td>44.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>147'06-0/0&quot;</td>
<td>348-04</td>
<td>08'05-0/0&quot;</td>
<td>24.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>147'06-0/0&quot;</td>
<td>168-51</td>
<td>08'08-1/4&quot;</td>
<td>44.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>147'06-0/0&quot;</td>
<td>191-56</td>
<td>08'05-0/0&quot;</td>
<td>24.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>143'00-1/4&quot;</td>
<td>354-16</td>
<td>08'04-3/4&quot;</td>
<td>60.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>143'00-1/4&quot;</td>
<td>004-13</td>
<td>08'03-0/0&quot;</td>
<td>60.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>143'00-1/4&quot;</td>
<td>185-44</td>
<td>08'04-3/4&quot;</td>
<td>60.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>143'00-1/4&quot;</td>
<td>175-47</td>
<td>08'03-0/0&quot;</td>
<td>60.000&quot;</td>
<td>1.500&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
<tr>
<td>162'00-0/0&quot;</td>
<td>348-51</td>
<td>05'10-0/0&quot;</td>
<td>24.000&quot;</td>
<td>1.250&quot;</td>
<td>SA-516 Gr-70</td>
<td>01.50 15.00</td>
</tr>
</tbody>
</table>

(1) Sleeve projection length on inside of containment wall
(2) Sleeve projection length on outside of containment wall
## Table 3.8-10

**Margin of Safety for Design of Containment Liner Plate and Anchorage System**

<table>
<thead>
<tr>
<th>Structure</th>
<th>Category</th>
<th>Allowable Value</th>
<th>Maximum Value</th>
<th>Ratio(^{(1)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>Liner Plate</td>
<td>Construction (stress)</td>
<td>20 ksi</td>
<td>16.4 ksi</td>
<td>1.22</td>
</tr>
<tr>
<td></td>
<td>Service (membrane strain)</td>
<td>0.002</td>
<td>0.0012</td>
<td>1.67</td>
</tr>
<tr>
<td></td>
<td>Service (membrane &amp; bending strain)</td>
<td>0.004</td>
<td>0.0042</td>
<td>0.95(^{(2)})</td>
</tr>
<tr>
<td></td>
<td>Factored (membrane strain)</td>
<td>0.005</td>
<td>0.0024</td>
<td>2.08</td>
</tr>
<tr>
<td></td>
<td>Factored (membrane &amp; bending strain)</td>
<td>0.014</td>
<td>0.0061</td>
<td>2.30</td>
</tr>
<tr>
<td>Liner Anchorage</td>
<td>0.25 delta u (displacement)</td>
<td>0.02285 in</td>
<td>0.00738 in</td>
<td>3.10</td>
</tr>
</tbody>
</table>

\(^{(1)}\) Ratio = Allowable Value / Maximum Value

\(^{(2)}\) This value is acceptable with the following two reasons: First, the anchor movement is based on the unbuckled panel having a 15% greater thickness than the buckled panel. This variation accounts for rolling tolerances of the 1/4" liner plate and it is considered conservative since it is very unlikely that the plate thickness fluctuations of this magnitude will occur within 30 inch plate width. Second, the creep and shrinkage strains at the end of the plant life have been combined with the initial prestress strains. This is too conservative because such a combination of maximum strains does not occur simultaneously.
### Computer Programs for Seismic Category I Structures

<table>
<thead>
<tr>
<th>Program</th>
<th>Analysis Method</th>
<th>Analysis Model</th>
<th>Analysis Scope</th>
<th>Validation &amp; Verification</th>
</tr>
</thead>
<tbody>
<tr>
<td>ANSYS</td>
<td>Modal analysis</td>
<td>Reactor containment building (shell &amp; dome, internal structure)</td>
<td>Eigenvalue analysis</td>
<td>ANSYS was procured with a Quality Assurance Service Agreement and meets the applicable requirements of the NQA-1, Subpart 2.7, quality assurance requirements of computer software.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Spent fuel pool &amp; aux. feed water storage tank</td>
<td>Eigenvalue analysis</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Response spectrum analysis</td>
<td>Reactor containment building (shell &amp; dome, internal structure)</td>
<td>Structural analysis of seismic load for RCB</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Static analysis</td>
<td>Reactor containment building (shell &amp; dome, internal structure)</td>
<td>Structural analysis of RCB for structure design (e.g., dead and live loads, etc.)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Auxiliary building (including spent fuel pool, aux. feed water storage tank)</td>
<td>Structural analysis of AB for structure design (e.g., dead and live loads, etc.)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Local analysis of spent fuel pool and aux. feed water storage tank (e.g., hydrostatic and hydrodynamic loads, etc.)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Emergency diesel generator building</td>
<td>Structural analysis of EDGB for structure design (e.g., dead and live loads, etc.)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Diesel fuel oil storage tank building</td>
<td>Structural analysis of DFOT for structure design (e.g., dead and live loads, etc.)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Heat transfer analysis</td>
<td>Reactor containment building (shell &amp; dome)</td>
<td>Temperature analysis</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Spent fuel pool</td>
<td>Temperature analysis</td>
<td></td>
</tr>
</tbody>
</table>
## Table 3.8-11 (2 of 3)

<table>
<thead>
<tr>
<th>Program</th>
<th>Analysis Method</th>
<th>Analysis Model</th>
<th>Analysis Scope</th>
<th>Validation &amp; Verification</th>
</tr>
</thead>
<tbody>
<tr>
<td>ANSYS (Cont.)</td>
<td>• Nonlinear analysis</td>
<td>• NI common basemat</td>
<td>• Structural analysis of basemat for structure design considering nonlinear soil spring (compressive only spring, reaction of superstructures)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Emergency diesel generator building basemat</td>
<td>• Structural analysis of EDGB for structure design considering nonlinear soil spring (compressive only spring, reaction of superstructures)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Diesel fuel oil storage tank building basemat</td>
<td>• Structural analysis of DFOT for structure design considering nonlinear soil spring (compressive only spring, reaction of superstructures)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Equivalent static analysis</td>
<td>• Auxiliary building (including spent fuel pool, aux. feed water storage tank)</td>
<td>• Structural analysis of seismic load for AB</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Emergency diesel generator building</td>
<td>• Structural analysis of seismic load for EDGB</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Diesel fuel oil storage tank building</td>
<td>• Structural analysis of seismic load for DFOT</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Direct integration time history analysis</td>
<td>• IRWST hydro-dynamic analysis</td>
<td>• Generation of floor response spectrum (FRS) due to POSRV sparger discharge load for mechanical and piping design</td>
<td></td>
</tr>
<tr>
<td>ABAQUS</td>
<td>• Nonlinear analysis</td>
<td>• Reactor containment building</td>
<td>• Ultimate pressure capacity evaluation corresponding to NRC RG 1.216, Position 1 using nonlinear material model • Combustible gas control inside containment evaluation corresponding to NRC RG 1.216, Position 2 using nonlinear material model</td>
<td>ABAQUS is validated in accordance with the registration procedure for computer software of KEPCO E&amp;C.</td>
</tr>
</tbody>
</table>
### Table 3.8-11 (3 of 3)

<table>
<thead>
<tr>
<th>Program</th>
<th>Analysis Method</th>
<th>Analysis Model</th>
<th>Analysis Scope</th>
<th>Validation &amp; Verification</th>
</tr>
</thead>
<tbody>
<tr>
<td>DARTEM</td>
<td>Static analysis</td>
<td>Reactor containment building (shell &amp; dome, internal structure, RCB basemat)</td>
<td>Structural analysis and design of reinforced concrete section subjected to mechanical and thermal loads</td>
<td>DARTEM is validated in accordance with the registration procedure for computer software of KEPCO E&amp;C.</td>
</tr>
<tr>
<td></td>
<td>Static analysis</td>
<td>Auxiliary building (Spent fuel pool)</td>
<td>Structural analysis and design of reinforced concrete section subjected to mechanical and thermal loads</td>
<td></td>
</tr>
<tr>
<td>LBAP</td>
<td>Static analysis</td>
<td>Reactor containment building (liner plate anchorage system)</td>
<td>Structural analysis of liner plate anchorage system when liner plate buckled</td>
<td>LBAP is validated in accordance with the registration procedure for computer software of KEPCO E&amp;C.</td>
</tr>
<tr>
<td>GTstrudl</td>
<td>Static analysis</td>
<td>Auxiliary building (concrete slab analysis model)</td>
<td>Structural analysis to obtain design forces for concrete slab design (e.g., dead and live loads, etc.)</td>
<td>GTstrudl was procured with a Quality Assurance Service Agreement and meets the applicable requirements of the NQA-1, Subpart 2.7, quality assurance requirements of computer software.</td>
</tr>
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</table>
### Table 3.8-12

**Maximum vertical displacement for construction and post-construction for NI, EDGB, and DFOT building**

a) Results of construction sequence analysis

<table>
<thead>
<tr>
<th>Structures</th>
<th>Category</th>
<th>Max. settlement</th>
<th>Soil profile S1</th>
<th>Soil profile S8</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>#1</td>
<td>#2</td>
</tr>
<tr>
<td>NI building</td>
<td>Construction (Sequence No. 58)</td>
<td>0.286</td>
<td>0.282</td>
<td>0.012</td>
</tr>
<tr>
<td></td>
<td>Post-construction (Sequence No. 59)</td>
<td>0.386</td>
<td>0.380</td>
<td>Not considered 1)</td>
</tr>
<tr>
<td>EDGB building</td>
<td>Construction</td>
<td>0.142</td>
<td></td>
<td>0.005</td>
</tr>
<tr>
<td></td>
<td>Post-construction</td>
<td>0.218</td>
<td></td>
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<td>0.266</td>
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1) Since soil profile S08 consists of rock profile, the creep effect of soil is not considered.

b) Summary of maximum vertical settlement criteria

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<td>Post-Construction</td>
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<tr>
<td>EDG building</td>
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<td>43.28mm (0.142ft)</td>
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<td>66.45mm (0.218ft)</td>
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<tr>
<td>DFOT building</td>
<td>Construction</td>
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<tr>
<td></td>
<td>Post-Construction</td>
<td>81.08mm (0.266ft)</td>
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Maximum tilting settlement for construction and post-construction for NI building

a) Results of construction sequence analysis

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1) Since soil profile S08 consists of rock profile, the creep effect of soil is not considered.

b) Summary of maximum tilting settlement criteria

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Differential settlement between structures
for all buildings under construction and post-construction

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<th>Post-construction Min.</th>
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<td>S08</td>
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1) Since soil profile S08 consists of rock profile, the creep effect of soil is not considered.
2) Maximum allowable differential settlement between buildings is 3 inch described in Table 2.0-1.
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Application of Loads based on Code Criteria

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APPENDIX 3.8A

STRUCTURAL DESIGN SUMMARY
# APPENDIX 3.8A – STRUCTURAL DESIGN SUMMARY

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APPENDIX 3.8A – STRUCTURAL DESIGN SUMMARY

This appendix provides the details of analysis and design for selected critical sections of seismic Category I structures. The critical design sections are the portions of safety-related, seismic Category I structures, which are credited in prevention or mitigation of consequences of postulated design basis accidents, expected to experience the largest structural demands during design basis conditions, or needed for safety evaluation of an essentially complete design.

To determine the critical design sections, structural types and materials such as concrete or steel, structural configurations representing locations and discontinuities are basically considered. Some selected critical sections may be typical of other portions of the structure, where the portions are not identified as critical sections due to their similarities with the selected design critical sections. In this case, the critical design sections are representative of an essentially complete structural design, and their design adequacy provides reasonable assurance of overall plant structural design.

Although certain portions are not subject to the limiting structural demands or can be considered less critical, they are necessary to be selected as critical sections due to their specific aspects such as design code and criteria. This may be a significant consideration because the structural demand based critical sections represent only those portions of a structure that experience high loads or stress and may not identify intervening structural elements that are not subject to high stress or loading but are needed for evaluating structural integrity.

In addition to structural features, safety-related functional role is also considered to select the critical design sections. Some of the APR1400 structures are required to achieve major performance whose failures could degrade system or equipment or pose safety hazard to plant personnel or to the general public.

The above criteria may be applied not only as one criterion, but also as mixed criteria based on engineering judgment and consistency. The specific contents for the critical design structures are presented in Subsection 3.8A.1 through 3.8A.3, and more detailed features in each structure can be further broken into portions, as described in each subsection. The design forces and moments in the tables of each subsection represent the governing set for the design of the structure.
3.8A.1 Reactor Containment Building

This section provides details of the analysis and design for the critical sections relevant to containment structures: containment wall and dome, internal structures, and basemat.

3.8A.1.1 Structural Description and Geometry

3.8A.1.1.1 Containment Wall, Dome and Basemat

The containment consists of a right circular cylinder closed on top by a hemispherical dome. The cylindrical wall, dome, and internal structures are supported on a nuclear island (NI) common basemat with a central cavity and tendon gallery. The cylindrical wall is anchored to the basemat by vertical reinforcements. The containment is constructed of concrete and prestressed by horizontal and vertical post-tensioned tendons in the wall and dome.

The basemat is constructed of conventional reinforced concrete.

An equipment hatch is provided for maintenance and removal of equipment (including the steam generators). Personnel airlocks are also provided.

The interior face of the reactor containment building is lined with 6 mm (0.25 in) thickness steel liner plate to form a leak-tight barrier and this liner plate is used as formwork during concrete placement. Thickened embedment plates and polar crane support brackets are welded into the liner plate. The basemat is lined with steel plate welded to embedded floor beams and covered by fill concrete.

3.8A.1.1.2 Containment Internal Structures

3.8A.1.1.2.1 Primary Shield Wall

The primary shield wall (PSW) is a reinforced concrete structure that houses the reactor, provides primary radiation shielding, and is an integral part of the internal structures. The PSW reinforcing is anchored into the containment basemat by the use of mechanical splices welded to both sides of a thickened liner plate. The PSW provides support for the reactor, refueling pool walls above the reactor cavity, and refueling pool slabs. The PSW forms a monolithic ring that surrounds the reactor vessel (RV). Penetrations in the PSW are provided for the primary loop piping and cavity ventilation system.
3.8A.1.1.2.2 In-Containment Refueling Water Storage Tank (IRWST)

The in-containment refueling water storage tank (IRWST) provides storage of refueling water, a single source of water for the safety injection and containment spray pumps, and a heat sink for discharges from the POSRVs and the reactor coolant gas vent system (RCGVS). The IRWST is annulus shaped and uses the lower section of the internal structure as its outer boundary. The IRWST is provided with a stainless steel liner to prevent leakage. The IRWST consists of a top and bottom slab and exterior and interior walls. The bottom slab of the IRWST rests on the containment basemat and the top and bottom slab are rigidly connected to the secondary shield wall (SSW).

3.8A.1.1.2.3 Secondary Shield Wall

The SSW is a circular reinforced concrete structure that, together with the refueling pool walls, encloses the nuclear steam supply system (NSSS) equipment, except for the reactor vessel. The SSW major reinforcing, which is necessary for carrying the seismic overturning moment, is anchored into the containment basemat in a manner similar to the PSW. The SSW provides lateral support for the steam generators, reactor coolant pumps (RCPs), and pressurizer and supports the operating and intermediate floors.

3.8A.1.2 Structural Materials

The principal construction materials for the reactor containment building are concrete, liner plate, reinforcing steel, and prestressing steel. The minimum specified material properties are used for the conservative design of the reactor containment building. The detailed material properties considered in the analysis and design of the reactor containment building are as follows:

3.8A.1.2.1 Concrete

The concrete used in the analysis and design of the reactor containment building is normal weight concrete.

The concrete characteristics of the containment wall, dome, and internal structures are as follows:

a. Compressive strength: 41.4 MPa (6,000 psi) at 91 days

b. Elastic modulus: 30,440 MPa (4,415 ksi)
c. Unit weight: 2,400 kg/m³ (150 pcf)

d. Poisson’s ratio: 0.17

The concrete characteristics of the containment basemat are as follows:

a. Compressive strength: 34.5 MPa (5,000 psi) at 91 days

b. Elastic modulus: 27,800 MPa (4,030 ksi)

c. Unit weight: 2,400 kg/m³ (150 pcf)

d. Poisson’s ratio: 0.17

3.8A.1.2.2 Liner Plate

The containment wall, dome, and basemat are lined with 6 mm (1/4 in) thick American Society of Mechanical Engineers (ASME) SA-516 and ASME SA-240 steel plates welded together to make a leak-tight membrane on the inside surface of the containment.

The characteristics of the carbon steel liner plate (ASME SA-516, Grade 60) are as follows:

a. Yield strength: 220 MPa (32 ksi)

b. Tensile strength: 415 MPa (60 ksi)

c. Elastic modulus: 200 GPa (29,000 ksi)

d. Poisson’s ratio: 0.3

The characteristics of the stainless steel liner plate (ASME SA-240, Type 304) are as follows:

a. Yield strength: 205 MPa (30 ksi)

b. Tensile strength: 515 MPa (75 ksi)

c. Elastic modulus: 200 GPa (29,000 ksi)

d. Poisson’s ratio: 0.3
3.8A.1.2.3 Reinforcing Steel

The material to be used for reinforcing bars for the containment conforms with American Society for Testing and Materials (ASTM) A615, Grade 60. The characteristics of the material are as follows:

a. Yield strength: 420 MPa (60 ksi)

b. Tensile strength: 620 MPa (90 ksi)

c. Elastic modulus: 200 GPa (29,000 ksi)

d. Poisson’s ratio: 0.3

3.8A.1.2.4 Prestressing Steel

The multi-strand tendons are used for the post-tensioning system of the containment wall and dome. The multi-strand system employs 15.24 mm (0.6 in.) diameter, seven-wire, low-relaxation strand manufactured in accordance with ASTM A416, Grade 270.

The characteristics of the tendons are as follows:

a. Elastic limit: 1,300 MPa (189 ksi)

b. Yield strength: 1,675 MPa (243 ksi)

c. Ultimate strength: 1,860 MPa (270 ksi)

d. Elastic modulus: 193 GPa (28,000 ksi)

e. Poisson’s ratio: 0.3

3.8A.1.3 Loads and Load Combinations

3.8A.1.3.1 Design Loads

The design loads applicable to the containment wall, dome, and basemat are in accordance with the requirements of CC-3200 of ASME Section III, Division 2, and U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.136. The design loads considered in the analysis and design of the containment wall, dome, and basemat are described in Subsection 3.8.1.3.2.
The design loads applicable to the containment internal structures are in accordance with the requirements of American Concrete Institute (ACI) 349. The typical design loads applicable to the containment internal structures are described in Subsection 3.8.4.3.

3.8A.1.3.2 Load Combinations

The design loads, applicable load factors, and load combinations for the analysis and design of the containment wall, dome, and basemat are in accordance with the requirements of CC-3200 of ASME Section III, Division 2, and NRC RG 1.136. The load combinations and applicable load factors considered in the analysis and design of the containment wall, dome, and basemat are shown in Table 3.8-2 (Chapter 3).

The design loads, applicable load factors, and load combinations for the analysis and design of the containment internal structures are in accordance with the requirements of ACI 349. The load combinations and applicable load factors considered in the analysis and design of the containment internal structures are shown in Table 3.8-7A (Chapter 3).

3.8A.1.4 Analysis and Design for Critical Sections

3.8A.1.4.1 Containment Wall and Dome

3.8A.1.4.1.1 Description

The containment encloses the entire pressurized water reactor vessel, steam generators, reactor coolant loops, pressurizer, and portions of the auxiliary and engineered safety features systems. The containment is a prestressed concrete structure composed of a cylindrical wall with a hemispherical dome and is supported by the NI concrete common basemat. The entire inside surface of the containment, including the basemat, is lined with a steel plate to provide reasonable assurance of the leak-tightness of the containment.

Access to the interior of the containment is provided through two personnel airlocks. An equipment hatch permits transfer of equipment into and out of the containment. Other major penetrations in the containment wall are those required for feedwater and main steam lines and various process pipe lines, electrical penetration assemblies, and the fuel transfer tube.

The cylindrical containment wall has a constant thickness of 1.37 m (4 ft 6 in.), starting from the top of the foundation basemat to the springline. The containment wall is
thickened locally around the equipment hatch, two personnel airlocks, and main steam line penetration areas to provide sufficient room for deflected prestressing tendons and additional heavy reinforcing bars. Three buttresses for anchoring of horizontal tendons are located at azimuths 90 degrees, 210 degrees, and 330 degrees and extend into the dome approximately 48 degrees above the springline. With the exception of the dome buttresses, the hemispherical dome has a constant thickness of 1.22 m (4 ft).

The cylindrical portion of the containment is prestressed by a post-tensioning system consisting of horizontal and inverted “U” vertical tendons. Each horizontal tendon is anchored at buttresses 240 degrees apart, bypassing the intermediate buttress. The dome portion of the containment is prestressed by a post-tensioning system consisting of horizontal tendons up to a 45-degree vertical angle and two groups of inverted “U” vertical tendons oriented 90 degrees to each other. The inverted “U” tendons are carried through the cylindrical wall and anchored at the tendon gallery in the common basemat.

3.8A.1.4.1.2 Load Combinations Considered

The following loading combinations are critical for the analysis and design of the containment wall and dome:

a. Test: 1.0 D + 1.0 L + 1.0 F_i + 1.0 P_t + 1.0 T_t
b. Construction: 1.0 D + 1.0 L + 1.0 F_i + 1.0 T_o + 1.0 W
c. Normal: 1.0 D + 1.0 L + 1.0 F_i + 1.0 T_o + 1.0 P_v
d. Severe: 1.0 D + 1.3 L + 1.0 F_r + 1.0 T_o + 1.0 W + 1.0 P_v
e. Extreme: 1.0 D + 1.0 L + 1.0 F_r + 1.0 T_o + 1.0 E_s + 1.0 P_v
f. Abnormal: 1.0 D + 1.0 L + 1.0 F_r + 1.5 P_a + 1.0 T_a
g. Abnormal/severe: 1.0 D + 1.0 L + 1.0 F_r + 1.25 P_a + 1.0 T_a + 1.25 W
h. Abnormal/extreme: 1.0 D + 1.0 L + 1.0 F_r + 1.0 P_a + 1.0 T_a + 1.0 E_s + 1.0 Y_r
3.8A.1.4.1.3 Analysis and Design Procedures

3.8A.1.4.1.3.1 General

The containment wall and dome are analyzed for the design loads and load combinations outlined in Subsections 3.8.1.3.2 and 3.8.1.3.3. Linear-elastic stress analyses are performed to investigate the structural behavior of the containment wall and dome using the general-purpose structural analysis program ANSYS.

The overall analysis of the containment wall and dome for various loading combinations is performed using a three-dimensional finite element (FE) model, considering large penetrations and locally thickened sections.

In order to consider the installation and testing of the polar crane before construction of the dome part, a separate FEM consisting only of the cylindrical wall is also constructed. The liner plate is not designed as a structural member to resist structural loads on the containment. It is, therefore, not included in the FEM.

The computer programs described in Section 3.8 are summarized in the Table 3.8A-40. Table 3.8A-40 shows the types of models, analysis methods, computer programs, and purposes of the structural analyses of the nuclear island structures, the emergency diesel generator building, and the diesel fuel oil storage tank building.

The acceptance criteria for the design of the containment wall and dome are defined based on the requirements in CC-3000 of ASME Section III, Division 2, and described in Subsection 3.8.1.5. Table 3.8A-1 shows the allowable stresses of concrete and reinforcing steel for service and factored loads, respectively.

3.8A.1.4.1.3.2 Analysis Model

Configuration of Containment Wall and Dome

The overall configuration of the containment wall and dome is shown in Figures 3.8-1 and 3.8-2. The representative dimensions of the containment wall and dome are as follows:

a. Inside diameter of the containment wall: 45.72 m (150.0 ft)

b. Inside height from the top of basemat to the dome apex: 76.66 m (251.5 ft)
c. Height from the top of the basemat to the springline: 53.80 m (176.5 ft)

d. Thickness of the containment wall: 1.37 m (4.5 ft)

e. Thickness of the containment dome: 1.22 m (4.0 ft)

An equipment hatch, two personnel airlocks, main steam line penetrations, and three buttresses are included in the analysis model. The locally thickened sections around the equipment hatch and personnel airlock are shown in Figure 3.8-3.

**Global Model**

The global model consists of solid elements for the concrete, truss elements for the tendon, and shell elements for the brackets of the polar crane. An eight-node, linear, solid element (SOLID185) in the ANSYS program is used to model the concrete part of containment wall and dome, including large thickened penetration areas and buttresses. Figure 3.8A-1 shows the schematic view of equipment hatch and personnel airlocks in the FEM. The basemat of the reactor containment building is included partially to consider the junction behavior and anchorage of the prestressing tendon. Figure 3.8A-2 shows the overall structural analysis model with FE mesh.

The prestressing tendon is modeled using truss elements, which have an axial degree of freedom only, and are embedded in the solid element part. A two-node, linear, three-dimensional truss element (LINK180) with initial tendon stress in the ANSYS program is used to model the prestressing tendon. Figure 3.8A-3 shows the tendon model using the truss elements.

The boundary condition is applied at the bottom of the basemat and vertically cut faces inside and outside of the basemat. The cut faces are fixed both radially and circumferentially, whereas the bottom face is fixed only vertically as is common when an unimportant part is cut for removal.

**Partial Model**

In order to consider the situation of installation and test operation of the polar crane before the dome is constructed, the partial model without the dome is constructed. This model is made simply by removing the solid elements above the El. 254 ft 6 in in the full model. The partial model is exactly the same as the full model, except for the dome part. The
boundary condition at the bottom of basemat is also the same as in the full model. Figure 3.8A-4 shows the shape of the partial model.

3.8A.1.4.1.3.3 Analysis Method

The analysis of the containment wall and dome is performed using the ANSYS program with a three-dimensional FEM. A brief description for the major loads considered in the analysis and design of the containment wall and dome are as follows:

Dead Load

The weight of concrete is automatically calculated from its mass density and gravitational acceleration by the ANSYS program. The additional weights such as those of the concrete slab, steel gratings, piping system, cable trays, heating, ventilation, and air conditioning (HVAC) systems, structural steels, and polar crane bracket are also considered dead loads.

Internal Pressure

The design basis accident (DBA) pressure of 413.7 kPa (60 psig) is applied to the inside surface of the containment wall and dome. The structural integrity test pressure is 1.15 times the design pressure. External or internal events such as containment spray actuation can induce a negative pressure on the containment. Therefore, the containment is designed for a negative pressure of 27.6 kPa (4 psig). To consider the pressure, which acts on the equipment hatch cover and attached doors in personnel airlocks, additional point loads are applied through reference nodes, which are located at the center of the penetration holes and inside surface of concrete wall. These reference nodes control and transfer the pressure loads to the surface of the corresponding cylindrical sleeve using the RBE3 command in the ANSYS program.

Prestress

The prestress is divided into two cases. The initial prestress is used for service categories to maximize the concrete compressive stress. For factored categories, the final (effective) prestress is considered to maximize the tensile stress of the reinforcing steels. The prestress losses are estimated in accordance with the guidelines specified in NRC RG 1.35.1. The stress profiles of vertical and horizontal tendons including losses of prestress are shown in Table 3.8A-41.
It is assumed that the stresses of the vertical tendon are constant from the anchor point to the springline and from the springline to the apex of the dome. Horizontal tendons that anchor to different buttresses 240 degrees apart are spaced closely enough to offset the stress variations. Hence, the stress, which is averaged throughout the length of the horizontal tendon, is used as a representative value.

**Temperature Load**

The thermal effect considers temperature variations during normal operating and accident conditions combined with the worst temperature condition (summer/winter) on the outside of the containment wall and dome. During normal operation, the containment is subject to a steady-state temperature condition. The linear gradient from the steady-state heat transfer analysis is applied to the stress analysis model. The containment is subject to a rapid temperature transient in the event of a loss-of-coolant accident (LOCA). The temperature transients result in a nonlinear temperature distribution within the concrete, which comes from the results of the transient heat transfer analysis.

**Operating Temperature**

- Inside (Liner Plate) : 120.0 °F
- Outside (exposed area) : 115.0 °F / -40.0 °F (Summer / Winter)
- Outside (enclosed area) : 104.0 °F / 50 °F (Max. / Min.)

**Accident Temperature**

- Inside (Liner Plate) : 290.0 °F
- Outside (exposed area) : 115.0 °F / -40.0 °F (Summer / Winter)
- Outside (enclosed area) : 104.0 °F / 50 °F (Max. / Min.)

**Seismic Load**

Structural analysis for the seismic load is based on the response spectrum analysis method, which computes the maximum response of a structure from the results of a modal analysis and their combinations. For the containment wall and dome, the in-structure response spectrum at El. 78 ft 0 in is used as a base excitation input. Figure 3.8A-5(b) shows the in-
structure response spectrum of the safe shutdown earthquake (SSE) level at El. 78 ft 0 in with 5 percent damping.

3.8A.1.4.1.3.4 Analysis Results

The section forces and moments for each element are calculated from the integration of stress resultants, which are obtained from various FE analysis results. The maximum section forces and moments for the principal design sections of the containment wall and dome are summarized in Table 3.8A-2.

3.8A.1.4.1.3.5 Design Sections

Critical sections are those portions of the containment wall and dome that (1) perform a safety-critical function, (2) are subjected to the largest stress demands, (3) are considered to be representative of the structural design, and (4) provide reasonable assurance that the structural design is being performed in a manner consistent with the guidance in the SRP, NRC RGs, and other regulatory requirements. Specific aspects of structures such as design code and criteria are also taken into account.

The sections at geometric discontinuities and the expected maximum stress location are considered as critical design sections. The major design sections for the containment wall and dome structures including liner plate/anchorage are as follows:

a. Base of containment wall (wall-basemat junction)

b. Mid-height of wall

c. Polar crane bracket level and springline

d. Thickened sections around large penetrations

   1) Equipment hatch
   2) Personnel airlocks

e. Containment dome

f. Containment liner plate/anchorage
3.8A.1.4.1.3.6 Rebar Arrangement

Continuous vertical and horizontal reinforcements are placed at the inside and outside faces of the containment wall. The vertical reinforcements of the containment wall are extended and anchored to the basemat. Additional reinforcing bars are provided around the large penetrations in the cylindrical wall as required. Shear ties are also provided where shear reinforcing is required.

Table 3.8A-3 summarizes the reinforcing details for the major design sections of the containment wall and dome. Figures 3.8A-6 through 3.8A-10, and 3.8A-53 show the rebar arrangement for the major design sections of the containment wall and dome. Figures 3.8A-16 and 3.8A-17 show the connection detail between the containment wall and basemat.

The containment dome is also reinforced by two-way orthogonal sets of vertical reinforcing steel and hoop reinforcing steel. The orthogonal reinforcing is the continuation of the vertical reinforcing in the containment wall. Hoop reinforcing is also provided up to 45 degrees above the springline. Radial reinforcements are provided over the entire dome to resist radial tension forces resulting from curved tendons.

3.8A.1.4.1.3.7 Design Results

The design sections of the containment wall and dome are analyzed by the computer program DARTEM to check the stresses of concrete and reinforcing steel in the concrete section. The input of DARTEM consists of section geometry, material properties, section forces and moments, and loading combinations. Table 3.8A-4 presents the rebar and concrete stresses and margins of safety for the major design sections of the containment. The margin of safety is the ratio of allowable stress and actual stress of rebar and concrete in the containment.

3.8A.1.4.1.3.8 Liner Plate and Anchorage

The containment liner plate primarily serves as a leak-tight barrier. It also acts as a form for concrete pouring of wall and dome during construction. The liner plate is anchored into the concrete by angle stiffeners. The angle stiffeners serve two major functions as an efficient means to give the liner plate bending stiffness and an increased load resistance during construction and as anchors to keep the liner plate from separating from the concrete during operating and accident conditions.
For these functions, the load combinations and related effects described in Subsections 3.8.1.3.4 and 3.8.1.4.10 are considered in the design of the liner plate and anchorage. The analysis, design, detailing and fabrication requirements of the liner plate and anchorage are performed in accordance with the requirements of ASME, Section III, Division 2, CC, Article CC-3600, CC-3700, CC-3800 and CC-4500, respectively.

According to the requirements of the ASME Code, the liner plate is designed to be within allowable strain criteria for service and factored conditions, and it is designed to be within allowable stress criteria for construction condition, as shown in Table 3.8A-45. The allowable capacity of liner anchors are specified in terms of allowable forces or displacement as shown in Table 3.8A-46.

The unbalanced forces resulting from variations in the liner curvature, liner thickness and liner strength are considered in the anchor analysis. The computer program LBAP is used to determine maximum anchor forces and displacements assuming the liner panel buckles.

For the structural design, the stress as a concrete form is calculated for basemat, shell, and dome liners. The results including the margin of safety for each liner plate/anchorage system are presented in Table 3.8-10.

3.8A.1.4.2 Containment Basemat

3.8A.1.4.2.1 Description

The NI common basemat structure of the APR1400 consists of the auxiliary building (AB) area base and reactor containment building (RCB) area base structure. The NI common basemat is a reinforced concrete mat foundation that covers an area with maximum dimensions of 106.0 m × 107.6 m (348 ft × 353 ft). The thickness of basemat is 10 ft (El. 45 ft 0 in through El. 55 ft 0 in.) in the AB area and variable thickness of 23 ft to 33 ft (El. 45 ft 0 in through El. 78 ft 0 in.) in the RCB area in general, except partial areas such as the tendon gallery and reactor cavity area. The top of the basemat in the RCB is at El. 78 ft 0 in, whereas the basemat is at El. 66 ft 0 in in the reactor cavity area. The arrangements of the structure are shown in Figure 3.8A-11.

The NI common basemat includes a hollow rectangular toroid for the tendon gallery. The tendon gallery provides a space to install and inspect the inverted "U" shaped vertical tendons. It is located entirely within the NI reinforced concrete basemat, as shown in Figures 3.8-1 and 3.8-2. The vertical tendons run up the cylindrical shell, over the dome in
a non-radial mesh pattern, run down to the tendon gallery on the opposite side, and are anchored at each end in the tendon gallery, as shown in Figure 3.8-4. As the tendon gallery is located entirely within the NI common basemat, it is analyzed and designed as a part of the common basemat. The codes and standards, loads and load combinations, design and analysis procedures, and structural materials for the tendon gallery are the same as those for the NI common basemat, and are described in Subsections 3.8.5.2 through 3.8.5.4, Subsections 3.8A.1.2.1 and 3.8A.1.2.3, and the following Subsections 3.8A.1.4.2.2 through 3.8A.1.4.2.4. For the analysis model, design section forces and design results of the NI common basemat, including the tendon gallery, are presented in Tables 3.8A-5 through 3.8A-13 and Figures 3.8A-13 through 3.8A-17.

The reactor containment basemat is reinforced at the top and bottom with layers of reinforcing steel bars. The reinforcing bars are arranged in the radial and hoop directions for top layers and in the orthogonal directions for bottom layers. The reinforcement at the upper portion of the tendon gallery is in the radial and hoop directions, and the reinforcement at the lower portion of the tendon gallery is in the rectangular pattern aligned with the plant NS and EW directions as shown in Figures 3.8A-16 and 3.8A-17, and Table 3.8A-12.

3.8A.1.4.2.2 Load Combinations Considered

The following loading combinations are critical for the analysis and design of the basemat:

a. Test: $1.0D + 1.0L + 1.0L_h + 1.0F + 1.0P_t$

b. Normal: $1.0D + 1.0L + 1.0L_h + 1.0F$

c. Severe: $1.0D + 1.3L + 1.3L_h + 1.0F$

d. Abnormal: $1.0D + 1.0L + 1.0L_h + 1.0F + 1.5P_a$

e. Abnormal/Extreme: $1.0D + 1.0L + 1.0L_h + 1.0F + 1.0P_a + 1.0Y_r + 1.0E_s$

3.8A.1.4.2.3 Analysis and Design Procedures

The design of the APR1400 adheres to a standardized design concept and can be constructed on various sites, including soil and rock sites. Among the eight soil profiles, three profiles (Soil profiles #1, #4, and #8) are considered as weak, moderate, and hard soil...
profiles for the NI common basemat based on comparison to subgrade moduli of each soil profile.

Although only three soil profiles (upper-, medium-, and lower-bound soil cases) are considered in the basemat structural analysis, the superstructure analysis results from enveloped seismic loading in 9 analysis cases are conservatively used in the basemat structural analysis. Therefore, it is concluded that the basemat structural analyses of the three soil profiles cover all of the basemat analyses of the soil categories given for the APR1400.

The load combinations for the basemat structure are summarized in Section 3.8A.1.4.2.2. A total of 288 load combinations (96 combinations × 3 soil profiles) were examined for the NI common basemat structure including phasing consideration of superstructures. The NI common basemat structure is analyzed using the ANSYS program FE computer model. The NI common basemat structure model includes the containment wall, dome, internal structures, AB, and common basemat foundation structure.

The FEM for superstructures, including the RCB wall and dome, RCB internal structure, and AB structure, are connected to the solid basemat model to simulate the stiffness effect of superstructures to the basemat. The FEM consists of a total of 317,373 nodes and 313,101 elements. Figure 3.8A-12 shows the full FEM for the basemat structural analysis. The AB structure, RCB internal structure, RCB wall and dome, and basemat structure FEMs are shown in Figure 3.8A-13.

In the case of soil stiffness, the distributed springs and soil finite elements are used for appropriate loading conditions. In the case of static loading, the distributed springs (LINK180) applied different subgrade moduli is used to consider boussinesq effect. The compression-only option was applied to the LINK180 elements of the ANSYS connected direction with the basemat structure, and the fixed boundary condition was applied to the other end side node of LINK180 element, as shown in Figure 3.8A-14. Axial (tributary) areas of each LINK180 element were calculated by applying unit pressure to the surface, which has the same geometry as the basemat model.

In the case of load combination including the seismic case, the soil finite elements are used to represent the overall dynamic foundation stiffness. In a manner consistent with the seismic SSSI analysis, the material for the foundation media model is calculated based on strain-compatible shear wave velocity from SASSI.
The reactions from the analysis results of the each superstructure except seismic load are used as nodal forces and moments to the basemat. However, seismic loads from superstructures and basemat are applied by using the equivalent static acceleration method to the basemat. Torsional load is separately considered in the separate basemat analysis. The results from this separate analysis are combined by the absolute sum method to the results from the seismic load analysis. Buoyancy loads (Lh) due to groundwater are applied to the bottom of the basemat structure. The probable maximum water level used for the buoyancy loads calculation is El. 100 ft 0 in (ground level) for added conservatism.

Both the linear case (fully connected basemat to foundation) and nonlinear case (no connectivity between basemat and foundation when basemat uplift occurs) are included in the design. Based on a comparison between member forces and between the nonlinear case and the SSI analysis, 96 cases are sufficient to encompass all permutations caused by the superstructure. Therefore, the conservative design of the basemat is performed under linear and nonlinear condition since it bounds the problem of no uplift/uplift. The envelop of these two cases is used for the design of the members. Under the nonlinear condition, 96 were performed cases using the 100-40-40 method, considering different phasing of three superstructures. A detailed description and comparison results are presented in Technical Report, APR1400-E-S-NR-14006-P (Reference 40 in Subsection 3.8.7). The analysis results are expressed as the normal stresses and the shear stresses of solid elements. The stresses of solid elements are filed with respect to the rectangular and cylindrical coordinate systems to fit with the arrangements of reinforcement.

To envelop the flexural and shear reinforcement for the 36 load combinations, the RCB basemat is divided into eight design sections as represented in Table 3.8A-5. Figure 3.8A-15 shows design sections for the containment basemat.

Tables 3.8A-6 through 3.8A-9 show the calculated section forces and moments for the design. The calculated design forces and moments are used as input in the concrete section design program DARTEM for the design of flexural reinforcement and shear reinforcement. The design of the concrete sections is based on the ASME Section III, Division 2.

3.8A.1.4.2.3.1 Design Summary

The results on the design of the flexural and shear reinforcement are summarized in Tables 3.8A-10 through 3.8A-13. For the flexural reinforcement, it is confirmed that the maximum stresses of the provided reinforcement do not exceed the allowable stresses for
both the service and factored load conditions. For the shear reinforcement, it is confirmed that the amounts of provided reinforcement are sufficient to meet the demands of the required reinforcement for each design section. The margins of safety of the flexural and shear reinforcement and concrete stresses are shown in Table 3.8A-10 and 3.8A-11. The design envelops the given parameters so that the design is adequate for any specific site conditions within those parameters. Figures 3.8A-16 and 3.8A-17 show the rebar arrangement for the basemat of the RCB.

3.8A.1.4.2.3.2 Stability Check

The NI common basemat structure is evaluated for stability against overturning, sliding, and flotation. The calculated factors of safety against overturning, sliding, and flotation for the load combinations meet the criteria of Section II of SRP 3.8.5 as shown in Table 3.8A-14.

The sliding and overturning factors of safety are determined using load combination containing dead load (D), wind load (W), SSE (Es), and buoyant load at normal (He). The flotation factor of safety is determined based on dead load (D) and buoyant force at flood (Hs).

For calculation of buoyant load at normal (Hc), the design groundwater level is applied, while the extreme groundwater level is applied to calculate the buoyant force at flood (Hs). The design groundwater level is El. 96 ft 8 in. The extreme groundwater level is the same as plant grade level (El. 98 ft 8 in.) considering probable maximum flood.

In the earthquake load, axial force, shear force, and moment due to horizontal and vertical excitation of the structure are obtained from seismic analysis. Since seismic load governs over tornado load, the tornado load is not necessary to be considered in the stability evaluation. The wind load combination (D + Hc + W) in Table 3.8A-14 is considered though the seismic load governs over the wind load, because the allowable factor of safety for wind is 1.5, which is different from 1.1 for the seismic load. A summary of overturning, sliding, and flotation check is provided in Table 3.8A-15.

3.8A.1.4.2.3.3 Basemat Uplift Check

The ground contact ratio between the basemat and soil is carried out to provide reasonable assurance that the linear soil-structure interaction (SSI) analysis remains valid. The ground contact ratio is defined as combining the stresses obtained from soil spring under static
loads with stresses obtained from time histories under seismic loads. Table 3.8A-16 shows the ground contact ratio of NI common basemat. The APR1400 NI common basemat has an 80 percent or more contact area during basemat uplift, and it can be concluded that the contact area would be acceptable.

3.8A.1.4.2.3.4 Settlement Check

Differential settlements are divided by the differential settlement within the NI common basemat and the differential settlement between the NI basemat and other seismic category I structures.

Figure 3.8A-18 shows the node location at the bottom of the NI common basemat for checking the settlement. The nodes within a distance of approximately 15.24 m (50 ft) are selected to check the different settlements. Table 3.8A-17 shows the differential settlements at site profiles 1, 4, and 8. The maximum differential settlements per 15.24 m (50 ft) at site profiles 1, 4, and 8 are 5.309 mm (0.209 in.), 2.311 mm (0.091 in.), and 0.991 mm (0.039 in.), respectively.

In addition, differential settlement between NI common basemat and other buildings basemat is evaluated for seismic category I structures. For the case of differential settlement between the NI common basemat and TGB basemat it is not evaluated since TGB is not a seismic category I structures and there are no safety related systems in TGB.

3.8A.1.4.2.3.5 Bearing Pressure

The bearing pressures of the NI common basemat is evaluated for soil profile S01 (weakest), S04 (moderate), S08 (strongest) under static and dynamic loading conditions. The model for the superstructure, NI common basemat, and soil used for the bearing pressure evaluation is identical as that used for the design described in 3.8A.1.4.2.3.

The maximum static bearing pressure is determined as the soil spring forces divided by the tributary area of the soil spring under the dead and live load conditions. The maximum dynamic bearing pressure is determined as the contact pressure between the basemat and foundation media model for statics plus seismic loads. The maximum static bearing pressure of 937.1kPa (19,570 lb/ft²) in the APR1400 NI common basemat is obtained from the basemat analysis results of soil profile S01. A value of 20ksf is defined as the allowable static bearing demand to provide an additional margin of safety at the site. The maximum dynamic bearing pressure of 2586.2kPa (54,010 lb/ft²) is obtained from the
basemat analysis results of soil profile S08. A value of 60ksf is defined as the allowable dynamic bearing demand to provide an additional margin of safety at the site.

3.8A.1.4.2.4 Conclusion

The basemat concrete section strengths determined from the ASME criteria are sufficient to resist the design basis loads. It is feasible to design and construct the structural components considered. The assumptions envelop the given parameters so that the design presented is adequate for any specific site conditions within those parameters.

3.8A.1.4.3 Internal Structures

3.8A.1.4.3.1 Primary Shield Wall

3.8A.1.4.3.1.1 Description

The PSW is a massive rectangular concrete structure, 18.80 m (61 ft 8 in.) long by 11.43 m (37 ft 6 in) wide, with cavities consisting of the following:

a. Vertical chase, 2.03 m (6 ft 8 in.) by 5.18 m (17 ft 0 in.), for in-core instrumentation (ICI) guide tubes from the seal table at the bottom of the refueling pool, El. 130 ft 0 in, down to the bottom of the ICI tunnel at El. 69 ft 0 in.

b. Horizontal chase, 5.51 m (18 ft 3/4 in.) wide 4.27 m (14 ft 0 in.) high, from below the seal table to below the reactor vessel for the ICI guide tubes.

c. A cavity to enclose and support the reactor vessel from the top of the PSW at El. 130 ft 0 in to the bottom of the ICI horizontal chase.

d. Openings to allow installation and access to the main coolant loop piping from the reactor vessel to the steam generators and the RCPs back to the reactor vessel.

e. A laydown area for the upper guide structure that is a part of the fuel handing system. This opening is 5.18 m (17 ft 0 in.) by 5.16 m (16 ft 11 in.) and extends from the bottom of the refueling pool down to El. 106 ft 6-3/8 in.

3.8A.1.4.3.1.2 Load Combinations Considered

The following loading combinations are critical for the analysis and design of the PSW:
a. Normal: $1.4D + 1.4L_h + 1.7L + 1.7R_o + 1.1D + 1.1L_h + 1.3L + 1.2T_o + 1.3R_o$

b. Abnormal: $1.0D + 1.0L_h + 1.0L + 1.4P_a + 1.0T_a + 1.0R_a$

c. Extreme environmental: $1.0D + 1.0L_h + 1.0L + 1.0T_o + 1.0E_s + 1.0R_o$

d. Abnormal/extreme environmental: $1.0D + 1.0L_h + 1.0L + 1.0P_a + 1.0T_a + 1.0R_a + 1.0Y_r + 1.0E_s + 1.0Y_j + 1.0Y_m$

Pipe break reaction ($Y_r$), jet impingement loads ($Y_j$), and missile impact loads ($Y_m$) do not act on the PSW. So, these loads are not applied to the design of the PSW.

3.8A.1.4.3.1.3 **Analysis Methods and Results**

The containment internal concrete structures are interconnected at various elevations. Significant lateral loads from the reactor coolant system (RCS) supports are applied at several elevations. In order to properly account for the load distribution in structures, an overall structural model representing containment internal concrete structures is prepared. Operating concrete floor slabs between the SSWs and containment shell are included as reaction forces obtained from local analyses to represent dynamic amplification in the finite element model (FEM).

The ANSYS program is used to perform structural analysis using the containment internal structure full model. The FEM consists of a total of 50,496 nodes. The numbers of shell, solid, and beam elements are 5,522, 41,689, and 827, respectively. The following containment internal structures are included in the analysis model:

**Solid Elements**

a. PSW

b. IRWST and fill concrete

**Shell Elements**

a. SSW

b. Refueling pool wall and slab

c. Pressurizer (PZR) enclosure wall and slab
d. Steam generator (SG) enclosure wall

e. Operating floor slab between SSW and refueling pool wall

**Frame Elements**

a. RCP lateral support beam

b. RCS model

Figure 3.8A-19 shows the full FEM for the containment internal structure. The solid element model (PSW, IRWST, and fill concrete), shell element model (SSW), and beam element model (RCS) are shown in Figure 3.8A-20.

Structure dead load consists of self-weight for PSW, SSW, RV, additional weight of floor and equipment, and dead load of RCS. Fifty percent of the weights and equipment weights on the floor between the containment shell and the SSW are assumed to be distributed to the containment shell and the SSW, respectively. The large equipment weights are applied as nodal forces at the location of equipment loads.

The live load is applied as follows:

a. Concrete slabs at El.100 ft 0 in and El.156 ft 0 in: 0.2 ksf
   (1.0 ksf for movable equipment during construction and maintenance)

b. Other slabs: 0.2 ksf

Hydrostatic loads are divided by the surface pressure loads in the refueling pool and IRWST walls and bottom slabs.

An equivalent uniform temperature gradient is input directly in the ANSYS model at the appropriate nodes. The temperature profiles during normal operating condition are more severe than those of the accident condition, thus represent the limiting temperature for all the plant conditions.

Compartment pressures on RCB internal structures are result of a pipe break inside containment. The types of compartment pressures are as follows:

a. SG compartment – feedwater economizer nozzle
b. SG compartment – feedwater downcomer nozzle  
c. SG compartment – SG blowdown nozzle  
d. PZR compartment – PZR spray nozzle  
e. PZR compartment – POSRV nozzle  
f. PZR spray valve room – PZR spray line

Branch line pipe break (BLPB) loads are dynamic reactions caused by the combined effects of branch line nozzle reactions or thrust due to pipe break, jet impingement on RCS equipment, or subcompartment pressure effects on RCS equipment. The RCS support reactions due to BLPB are applied as nodal forces at the support locations.

The hydrodynamic pressure load, which is generated by the expulsion of air in the pilot-operated safety relief valve (POSRV) discharge, is applied to the wall and bottom slab of the IRWST through the twelve spargers. For the hydrodynamic pressure load, by multiplying the dynamic impact factor (DIF), the maximum pressure is conservatively considered as the static load in the analysis. In addition, the normalized factor is considered for the spatial distribution due to the location of spargers.

The seismic analysis for structures is performed using response spectrum analysis. A 7 percent damping ratio for reinforced concrete structures (SSE) and 3 percent damping ratio for the RCS model are used. In addition, the damping ratio for the IRWST or refueling pool is the same as that for reinforced concrete structures: the seismic response of water is only considered as impulsive (rigid) mode for structural analysis. Figure 3.8A-5 (a) and (c) show the in-structure response spectrum (ISRS) of the SSE level at El. 78 ft 0 in with 3 percent and 7 percent damping.

Three sections are selected in the PSW as critical sections. Each section is thinnest in the directions of north, south, and east. The design forces and moments for PSW critical sections are presented in the Table 3.8A-18. Table 3.8A-22 presents the margins of safety of reinforcement in the primary shield wall. The margin of safety is the ratio of provided reinforcement and required reinforcement.
3.8A.1.4.3.1.4 Typical Rebar Arrangement

The typical rebar arrangements for PSW are presented in the Table 3.8A-21. Figure 3.8A-58 shows the rebar arrangement for major design section of the PSW.

3.8A.1.4.3.1.5 Conclusion

The PSW concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads. It is feasible to design and construct the structural components that are considered. The assumptions envelop the given parameters so the design is adequate for any site-specific conditions within the parameters.

3.8A.1.4.3.2 IRWST

3.8A.1.4.3.2.1 Description

The IRWST provides storage of refueling water, a single source of water for the safety injection and containment spray pumps, and a heat sink for discharges from the POSRVs and the RCGVS. The IRWST is annulus shaped and uses the lower section of the internal structure as its outer boundary. A stainless steel liner is provided inside the IRWST.

The IRWST consists of a top and bottom slab and exterior and interior walls. The bottom slab rests on the containment basemat and the top and bottom slabs are rigidly connected to the SSW. The interior wall is the reactor containment SSW. The fill concrete portion is placed between the SSW and PSW from El. 78 ft 0 in to El. 100 ft 0 in.

The exterior wall thickness of the IRWST is 0.91 m (3 ft), the same as the bottom and roof slabs. The inner radius of the exterior wall is 21.89 m (71 ft 10 in.) and the outer radius of the interior wall (SSW) is 16.15 m (53 ft 0 in.). The top elevation of the IRWST bottom slab is El. 81 ft 0 in. The top elevation of the IRWST roof slab is El. 100 ft 0 in. The normal water level in the IRWST is at El. 93 ft 0 in.

3.8A.1.4.3.2.2 Load Combinations Considered

The following loading combinations are critical for the analysis and design of the IRWST wall:

a. Normal: \(1.4D + 1.4L_h + 1.7L + 1.7R_o + 1.7P_{ac}\) and \(1.1D + 1.1L_h + 1.3L + 1.2T_o + 1.3R_o + 1.3P_{ac}\)
b. Abnormal: \(1.0D + 1.0L_h + 1.0L + 1.5P_a + 1.25P_{ac} + 1.0T_a + 1.0R_a\)

c. Extreme environmental: \(1.0D + 1.0L_h + 1.0L + 1.0T_o + 1.0E_s + 1.0R_o + 1.0P_{ac}\)

d. Abnormal/extreme environmental: \(1.0D + 1.0L_h + 1.0L + 1.0P_{ac} + 1.0P_a + 1.0T_a + 1.0R_a + 1.0Y_r + 1.0Y_j + 1.0Y_m + 1.0E_s\)

e. Severe environmental: \(1.4D + 1.4L_h + 1.7L + 1.7R_o + 1.7P_{ac}\) and \(1.1D + 1.1L_h + 1.3L + 1.2T_o + 1.3R_o + 1.3P_{ac}\)

Accident pressure (\(P_a\)), Pipe break reaction (\(Y_r\)), jet impingement loads (\(Y_j\)), and missile impact loads (\(Y_m\)) do not act on the IRWST. So, these loads are not applied to the design of the IRWST.

\(P_{ac}\) is the air-clearing load, which is the hydrodynamic load generated by the expulsion of air in POSRV discharge lines during the POSRV discharge following the water clearing phenomena in the sparger.

3.8A.1.4.3.2.3 Analysis Methods and Results

The IRWST FEM is part of the containment internal structure full model. See Subsection 3.8A.1.4.3.1.3. The governing load to the IRWST outer wall and upper slab is the sparger discharge load. Hydrodynamic loads occur at twelve sparger locations (north and west). Because the IRWST is separated from the containment wall by a seismic gap and there is no connection to transfer POSRV load to the containment wall, the POSRV load is not directly applied on the containment wall. The primary and secondary shield wall (PSW and SSW) may be directly influenced by the POSRV load because this load is applied to the IRWST which is integrally connected with the PSW and SSW. However, the transient displacements of PSW and SSW due to POSRV load are comparatively small enough to be ignored. In addition, the spectral acceleration of POSRV is obtained from FRS curves which are generated from time-history analysis. The results comparing with seismic load show that the spectral acceleration by POSRV is less enough to be ignored for PSW and SSW. For reference, the developed FRS is used by other disciplines for qualification of systems and components (e.g., pipes and equipment). Therefore, stresses on the portions of outer wall and upper slab are investigated and critical sections are selected where the largest stress takes place. The design forces and moments for IRWST critical sections are presented in Table 3.8A-19. Table 3.8A-42 presents the margins of safety of reinforcement.
in the IRWST. The margin of safety is the ratio of provided reinforcement and required reinforcement.

The typical rebar arrangements for the IRWST are presented in the Table 3.8A-23.

3.8A.1.4.3.2.4 Conclusion

The IRWST wall/slab concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads. It is feasible to design and construct the structural components considered. The assumptions envelop the given parameters so the design is adequate for any site-specific conditions within the parameters.

3.8A.1.4.3.3 SSW

3.8A.1.4.3.3.1 Description

The SSW is a circular reinforced concrete structure that extends up to the main operating floor. The wall is anchored to the containment basemat. The wall is integrally connected to the refueling pool at the fuel transfer tube side (east) and at the regenerative heat exchanger room (west). At other points, the SSW is connected to the refueling pool through the SG and PZR enclosure walls and RCP lateral support members, which make the SSW and the internal structures almost symmetric around the east-west centerline of the containment.

In addition to enclosing the primary loop and the internal structures, the SSW provides lateral support for the SGs, RCPs, PZR, and the operating and intermediate floors inside the containment.

The major floor elevations are as follows:

a. Base floor: El. 100 ft 0 in
b. Intermediate floor: El. 114 ft 0 in, El. 136 ft 6 in
c. Operating floor: El. 156 ft 0 in

The major design dimensions of the secondary shield wall are as follows:

a. Wall thickness: 1.22 m (4 ft)
b. Inside radius: 14.94 m (49 ft)

c. Height of wall: 17.07 m (56 ft)

3.8A.1.4.3.3.2 Load Combinations Considered

The following loading combinations are critical for the analysis and design of the SSW:

a. Normal: $1.4D + 1.4L_h + 1.7L + 1.7R_o$ and $1.1D + 1.1L_h + 1.3L + 1.2T_o + 1.3R_o$

b. Abnormal: $1.0D + 1.0L_h + 1.0L + 1.4P_a + 1.0T_a + 1.0R_a$

c. Extreme environmental: $1.0D + 1.0L_h + 1.0L + 1.0T_o + 1.0R_o + 1.0E_s$

d. Abnormal/extreme environmental: $1.0D + 1.0L_h + 1.0L + 1.0P_a + 1.0T_a + 1.0R_a + 1.0Y_r + 1.0Y_j + 1.0Y_m + 1.0E_s$

Jet impingement loads ($Y_j$), missile impact loads ($Y_m$), reactions of pipe, cable tray and duct ($R_o$), and pipe accident reactions ($R_a$) are evaluated in local affected area. Thermal operating load ($T_o$), and accident temperature ($T_a$) do not affect the design of SSW since there is no temperature difference between the inside and outside of the SSW.

3.8A.1.4.3.3.3 Analysis Methods and Results

The SSW FEM is a part of the containment internal structure full model. See Subsection 3.8A.1.4.3.1.3. The SSWs extend from El. 100 ft 0 in up to the operating floor at El. 156 ft 0 in. The SSW from El. 100 ft 0 in to El. 114 ft 0 in is selected as the critical section because this portion of the wall includes the junction between SSW and fill concrete.

The refueling pool walls extend from the bottom of the pool at El. 130 ft 0 in up to El. 156 ft 0 in. The north, south, and west walls between these elevations are selected as critical sections.

SG enclosure walls extend from El. 156 ft 0 in up to El. 191 ft 0 in, which is the top of wall. SG enclosure walls between these elevations are selected as critical sections.

PZR enclosure walls extend from El. 133 ft 4 in up to El. 200 ft 0 in, which is the top of wall. PZR enclosure walls from El. 156 ft 0 in up to 191 ft 0 in are selected as critical sections since these portions of the wall support the PZR laterally. The slab at El. 156 ft 0
in is selected as the critical section because the vertical g-value at this elevation which is used for slab design is bigger than the g-value at other elevation.

The design forces and moments for SSW critical sections except the slab are presented in the Table 3.8A-20. Table 3.8A-25 presents the margins of safety of reinforcement in secondary shield wall except the slab. The margin of safety is the ratio of provided reinforcement and required reinforcement. The design forces and moments for the slab are presented in the Table 3.8A-43. Table 3.8A-44 presents the margins of safety of reinforcement at the critical section for the slab. The margin of safety is the ratio of provided reinforcement and required reinforcement.

3.8A.1.4.3.3.4 Typical Rebar Arrangement

The typical rebar arrangements for the SSW and the slab at El. 156 ft 0 in are presented in the Table 3.8A-24 and Table 3.8A-44, respectively. Figure 3.8A-57 shows the rebar arrangement for the slab at El. 156 ft 0 in and Figures 3.8A-59 through 3.8A-62 show the rebar arrangement for each design section of the SSW.

3.8A.1.4.3.3.5 Conclusion

The SSW concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads. It is feasible to design and construct the structural components considered. The assumptions envelop the given parameters so that the design presented is adequate for any specific site conditions within those parameters.

3.8A.1.4.3.4 Structural Steel Beam

3.8A.1.4.3.4.1 Description

In the RCB, there is no steel column structure. For steel beam structure, there are three major elevations in the annulus area of the RCB. The steel beams are located between containment wall and secondary shield wall at El. 114 ft 0 in, El. 136 ft 6 in, and El. 156 ft 0 in supporting the concrete slabs and grating area. Typical steel beam, beam connection, and beam seat on each level are designed for highest load case. The concrete slab and steel beams are analyzed separately. The loads on the slab and slab weight are applied on the steel beam by line load.
3.8A.1.4.3.4.2 Load Combinations Considered

In sixteen load combinations given in Table 3.8-7B, only the governing load cases are considered as defined herein. The load case 5 (LC5) governs over the construction conditions (LC1, LC2, LC3), the test condition (LC4), and the normal condition (LC6), and the severe environmental conditions (LC7, LC8, LC9). The load case 13 (LC13) governs over the abnormal conditions (LC10, LC11, LC12), extreme environmental conditions (LC14, LC15) and abnormal/extreme environmental conditions (LC16). The load of $T_o$, $T_a$ are not considered since the sliding connections are provided in the steel design to relieve the thermal stresses. The loads of $W$, $H$ are not considered since the loads do not exist inside of the containment. The loads of $P_a$, $R_a$, $Y_r$, $Y_j$, $Y_m$, $Y_f$, $M_a$, $W_t$, $H_s$ are not considered since the loads do not affect the steel structures. The load combinations considered are used as input for the analysis by computer program, and they are investigated for all structural members.

3.8A.1.4.3.4.3 Analysis and Design Methods

The computer program GTSTRUDL, which is used for structural analysis, is a software for creation of model, modification of the model, execution of analysis, check of the analysis, and optimization of design. The steel beam structures for design load cases are analyzed using 3-dimensional frame elements. The GTSTRUDL prints the detailed output of results including the stress. For design of steel beam, uniform floor loads are distributed to all beams based on tributary areas. The span direction for concrete slabs on metal deck is considered in determining the tributary areas. Response spectrum analysis is used to design steel member with the FRS that envelop the containment side and SSW side at each level. The steel beam model is built and analyzed for static and seismic loads utilizing GTSTRUDL. After the analysis, the stresses of steel structure are checked according to the allowable values in AISC N690. The allowable stresses in AISC N690 are used for stress acceptance criteria.

Connections are designed based on the reactions from the GTSTRUDL analysis. The capacities of the various connection components are computed. For design, the SSW side connection is composed of a beam seat and welding between the beam and beam seat. The beam seat and stiffener support downward vertical load and the welding between the beam and beam seat support upward vertical load. The axial and horizontal load and torsional moment are supported by concrete slab.
The sliding connection at the containment wall is composed of a beam seat, lower key bumper and gaps (between the end of the steel beam and the containment wall, between the lower key bumper and beam seat). The gap between the end of the steel beam and the containment wall is to allow radial movement. The other gap between the lower key bumper and beam seat is to allow horizontal movement due to seismic and thermal loads. The beam seat support downward vertical load and the lower key bumper support upward vertical load.

Friction force are considered for steel connection design. The friction coefficient of 0.35 is used as recommended in Commentary on the AISC 360-05 chapter J and the type of steel surface is class A.

3.8A.1.4.3.4.4 Conclusion

The design of steel beam and connections is performed to maintain adequate design margins. The details of the steel beam, concrete slab, and connection are shown in Figure 3.8A-56. The summary of design results is shown in Table 3.8A-47.

3.8A.2 Auxiliary Building

This section provides details of the analysis and design for the critical sections relevant to the auxiliary building (AB) structures: reinforced concrete shear wall, slab, concrete frame, and basemat.

3.8A.2.1 Structural Description and Geometry

The auxiliary building consists of the main control room (MCR) area, electrical and control area, main steam valve house area, chemical and volume control system (CVCS) area, emergency diesel generator (EDG) area, and fuel handling area. The building description is provided in Subsection 3.8.4.1.1.

The auxiliary building is a seismic Category I reinforced concrete structure, which is composed of rectangular walls, floor slabs, concrete columns, and beams. The AB surrounds the reactor containment building (RCB) and shares a common foundation basemat with the RCB. Both the structural design and physical arrangement of the AB provide protection against both external and internal hazards.
The slabs and shear walls in the building represent the primary lateral and vertical load-resisting system and are designed for both gravity and seismic-related loads. Concrete slabs at various elevations in the building distribute lateral forces (through diaphragm action) to the shear walls as in-plane loads, and resist vertical forces (self-weight and seismic forces) as out-of-plane loads.

Lateral loads are transferred down to the basemat foundation through shear walls as in-plane shear forces and moments. Vertical loads on slabs are supported either by concrete beams or walls. Those are transferred to the basemat foundation by the walls and the frames composed of concrete columns and beams.

3.8A.2.2 Structural Materials

The major materials that are used in the construction are concrete, reinforcing bars, and structural steels.

Concrete

The minimum concrete compressive strength of the AB is 34.5 MPa (5,000 psi) at 91 days. The properties of the concrete are as follows:

a. Compressive strength: 34.5 MPa (5,000 psi) at 91 days

b. Elastic modulus: 27,800 MPa (4,030 ksi)

c. Unit weight: 2,400 kg/m³ (150 pcf)

d. Poisson’s ratio: 0.17

Reinforcing Bars

The reinforcing bars used in the AB are in accordance with ASTM, A615 Gr. 60. The properties of the reinforcing bars are as follows:

a. Yield strength: 420 MPa (60 ksi)

b. Tensile strength: 620 MPa (90 ksi)
3.8A.2.3 **Loads and Load Combinations**

3.8A.2.3.1 **Design Loads**

The following are major design loads that are considered in the design of the AB.

**Dead Load (D)**

Dead load includes the weight of structures such as slabs, roofs, decking, framing (beams, columns, bracing, and walls), and the weight of permanently attached major equipment such as tanks, machinery, and cranes. Equipment loads heavier than or equal to 44.5 kN (10 kips) are considered concentrated weights.

The attachment loads include equipment loads lower than 10 kips such as piping, cable tray, and HVAC loads. The minimum attachment loads are as follows:

- a. Concrete floors: 9.6 kN/m² (200 psf)
- b. Roof floors: 7.2 kN/m² (150 psf)
- c. Interior wall: 1.0 kN/m² (20 psf)
- d. Exterior wall: 0.5 kN/m² (10 psf)

**Live Load (L)**

Live loads are conventional floor loads to account for occupancy, maintenance, equipment removal, equipment laydown, and other loads that vary in intensity. The minimum live loads are as follows:

- a. Concrete floors: 9.6 kN/m² (200 psf)
- b. Roof floors: 4.8 kN/m² (100 psf)
- c. Equipment removal aisle: 23.9 kN/m² (500 psf)
- d. Seismic live load: 2.4 kN/m² (50 psf) (considered in seismic loading combination)

In order to account for wall attachments, shear walls are designed considering a minimum out-of-plane live load of 16.6 kN/m² (350 psf).
Soil and surcharge load ($L_g$) and hydrostatic load ($L_h$) are considered as live load.

**Wind Load (W)**

Wind load is the equivalent static load generated by the design wind velocity and is calculated in accordance with ASCE 7. The AB is designed for a 100-year recurrence interval wind.

**Safe Shutdown Earthquake Load ($E_s$)**

In the structural analysis for the AB, seismic loads are considered with the equivalent static load method involving equivalent horizontal and vertical static forces.

**Accident Pressure Load ($P_a$)**

Accident pressure is applied external or internal air, gas, or liquid pressure loads during abnormal operating conditions.

**High-Energy Line Break (HELB) Load ($Y_r, Y_j, Y_m, Y_f$)**

There are several high- and moderate-energy pipe lines routed through the AB. During an abnormal condition at the plant, it is postulated that these lines can break at various locations. The structural loads associated with pipe breaks include pipe break reactions ($Y_r$), jet impingement load ($Y_j$), missile impact load ($Y_m$), compartmental flooding load ($Y_f$), and compartment pressure ($P_a$).

3.8A.2.3.2 **Load Combinations**

The load combinations are addressed in Subsection 3.8.4.3.6 and are used for analysis and design of the AB and associated components.

3.8A.2.4 **Analysis and Design for Critical Sections**

This section summarizes the analysis and design for critical sections of the AB. The critical sections are listed below. The description of critical section, analysis and design methods, and design summary are provided for each critical section. The locations of critical sections are shown in Figures 3.8A-21 through 3.8A-24.

a. Basemat
1) AB area of the nuclear island (NI) common basemat

b. Shear walls
   1) North wall of the north main steam isolation valve (MSIV) house
   2) North wall of the north auxiliary feedwater storage tank (AFWST)
   3) West wall of the MCR
   4) West wall of the spent fuel pool (SFP)
   5) East wall of the fuel handling area (FHA)

c. Slabs
   1) Floor slab of the EDG Room at El. 100 ft 0 inch
   2) Pool bottom slab of the SFP at El. 113 ft 0 inch
   3) Floor slab below the main steam enclosure at El. 137 ft 6 inch

3.8A.2.4.1 Basemat

Description

The AB shares a common foundation basemat with the RCB. The foundation of the RCB and AB is a reinforced concrete mat structure with the maximum dimensions of 106.0 m × 107.6 m (348 ft × 353 ft). The thickness of the basemat is 3.05 m (10 ft) in the AB area. The bottom of the basemat is located at El. 40 ft 0 in and 45 ft 0 in, below the finished grade elevation. The AB basemat is reinforced at the top and bottom with layers of reinforcing steel bars. The reinforcing bars are arranged in the orthogonal directions for top and bottom layers.

Analysis and Design Methods

The reinforced concrete common basemat is analyzed and designed for the reactions due to static, seismic, and all other loads that affect basemat analysis at the base of the superstructures. The forces and moments of the basemat area are obtained by application of design loads to the integrated static three-dimensional FEM. The analysis methods and
procedures of the NI common basemat are described in Subsection 3.8A.1.4.2.3. Figure 3.8A-25 shows the solid element model of the basemat for the AB and RCB.

Since the rigid connection between walls and basemat is not simulated in the analysis model of the NI common basemat structure, the basemat of the AB is not subject to any moment that might occur. Therefore, the additional structural analysis is executed to obtain the magnitude of the moment transferred from walls and columns, which are subjected to lateral loads. The design forces and moments for the AB basemat are summarized in Table 3.8A-26.

The AB basemat model is divided into 15 element sets as shown in Table 3.8A-26. The design loads are sorted by these element sets and the basemat design is performed in each element set. The AB basemat is designed in accordance with the requirements of ACI 349. Required reinforcements are calculated based on the governing required capacities obtained from finite element analysis for each design area.

Top and bottom reinforcements are calculated considering axial force, out-of-plane flexural force, and in-plane shear force. The required reinforcements for axial force and out-of-plane flexural force are determined per ACI 349, but a methodology taken from the ACI handbook (ACI 340R) is adopted to combine axial force and out-of-plane flexural force. The required reinforcement for in-plane shear force is determined per ACI 349.

The resultant shear forces are checked with the design nominal concrete strength and appropriate shear reinforcement steel is provided in the areas where the resultant shear force exceeds the design nominal shear strength.

**Design Summary**

The basemat is symmetrically reinforced to resist the potential moments as a result of differential settlement of the foundation. The maximum top and bottom reinforcements are 3-#18 bar at 300 mm (12 in.) spacing for each direction. The maximum shear reinforcements are required as 2-#8 bar at 300 mm (12 in.) spacing around the RCB and #6 bar at 300 mm (12 in.) spacing provided to the other area. The required reinforcement summary and margins of safety are shown in Table 3.8A-27. The margin of safety is the ratio of required reinforcement and provided reinforcement.
The reinforcement arrangement is shown in Figure 3.8A-32 and the typical sections for the AB basemat are shown in Figures 3.8A-33 and 3.8A-34. Typical details of the basemat are shown on Figure 3.8A-35.

The concrete basemat strength determined from the criteria in Section 3.8.5 is sufficient to resist the design basis loads. It is feasible to design and construct the NI foundation basemat. The design envelops the given parameters so that the design presented is adequate for any specific site conditions, within those parameters.

3.8A.2.4.2 Shear Walls

Description

The AB shear walls, together with the diaphragm slabs, are the primary lateral load resistance system, and are designed essentially for both the seismic and the extreme wind loads.

Concrete slabs at the different elevations of the building distribute the lateral forces through diaphragm action as in-plane loads to the shear walls in proportion to the relative stiffness of the shear walls. These in-plane loads are transferred down to the basemat foundation in the form of in-plane shear forces and moments in the shear walls.

The in-plane shear forces and moments, which are obtained from the structural analysis of the AB, are combined with the applicable out-of-plane loads and vertical loads to determine the arrangement of steel reinforcements required for the AB shear walls.

Analysis and Design Methods

The AB shear walls are modeled and analyzed using finite element analysis. The global structural static analysis is performed using the ANSYS program to evaluate distribution of forces to various elements of the structure. Shell 181 is adopted for the shear wall element, which is suitable for analyzing thin to moderately thick shell structures. It is a four-node element with 6 degrees of freedom at each node. The 3-D view (toward -XY) of the FEM model for the AB is shown Figure 3.8A-26.

The element forces of the AB shear wall are calculated in each level, individual shear wall and all loading combinations, and these are used for shear wall design. The main output
data on shell elements by the ANSYS program are composed of in-plane axial forces, in- 
plane shear forces, out-of-plane bending moments, and transverse shear forces.

The AB shear walls are analyzed and designed in accordance with the requirements of ACI 
349. The required reinforcements are designed by the following procedure:

a. Design for in-plane shear forces

b. Design for out-of-plane bending moments with axial forces

c. Combine the required reinforcement calculated above procedure

d. Check for shear capacity: in-plane shear forces, out-of-plane shear forces, and 
shear friction forces between wall and floor.

Out-of-plane forces and moments are considered in the shear wall design, which are 
determined by hand calculations or local structural analysis as well as the global analysis. 
Out-of-plane loads include attachment loads and seismic loads such as the associated 
hydrodynamic loads and dynamic incremental soil pressure.

The exterior walls below the grade are designed to resist the worst-case lateral earth 
pressure loads (static and dynamic), soil surcharge loads, and loads due to groundwater 
(static and dynamic). Lateral earth pressure is equal to the summation of the static earth 
pressure plus the dynamic earth pressure. The dynamic earth pressures are determined as 
the governing case between the dynamic earth pressures calculated in accordance with 
ASCE 4-98 (Reference 47 in Subsection 3.8.7), Section 3.5.3, Figure 3.5-1, “Variation of 
Normal Dynamic Soil Pressure for the Elastic Solution” and those calculated based on the 
results of the SSI/SSSI analyses. The hydrodynamic effect of pure water is determined 
based on the hydrodynamic formula suggested by Matuo and O’Hara, Section 8.10, 
Equation 8.40 “Principles of Soil Dynamics,” Braja M.Das (Reference 48 in Subsection 
3.8.7). Design forces and moments are summarized in Table 3.8A-28 for the critical 
sections.

The local structural analysis is performed for the tanks in the AB such as the AFWT and 
SFP. Local loads such as hydrodynamic and hydrostatic pressures loads or thermal loads 
acting on the walls are considered in the analysis. The hydrodynamic analysis includes two 
horizontal modes and one vertical mode of combined fluid-tank vibration. The horizontal
pressure calculations consider the horizontal sloshing convective mode and the impulsive mode in which a portion of the water moves as a mass body along with the tank wall.

For the SFP, thermal loads for the state of the abnormal operation are considered to be 82.2 °C (180 °F) at the inside and 21.1 °C (70 °F) at the outside. The structural integrity of the wall sections for thermal load is evaluated by using the section analysis program DARTEM.

The design of the auxiliary shear wall structure is based on the strength design method specified in ACI 349 with the supplement requirements of NRC RG 1.142.

**Selection of Critical Sections**

The typical design for the shear wall of the AB uses the procedure described above except that the critical sections of the shear wall meet the more stringent requirement to provide reasonable assurance of the function and safety of the system on the shear wall or the potential load induced by abnormal and extreme environmental conditions. Critical sections are selected according to the following criteria:

- a. The section contains the primary safety-related system
- b. The section considers the specific load case under abnormal and extreme environmental conditions
- c. The section meets protection requirements such as aircraft impact or missile barriers

The north wall of the north MSIV house is designed as a safety-related structure to provide reasonable assurance for the safety of the internal systems. The design forces of this shear wall include the compartment pressure due to a pipe break in the system. As an exterior wall, the design of this shear wall considers the aircraft impact analysis results and meets the missile barrier requirements.

The north wall of the north AFWST is designed as a safety-related structure to contain auxiliary feedwater under specified conditions. The design forces for this shear wall include hydrostatic load and hydrodynamic loads induced by the water in the AFWST.

The west wall of the MCR is designed as a safety-related structure to provide reasonable assurance of the safety of the internal systems and to control the whole system under any
specified conditions. As an exterior wall, the design of this shear wall meets the missile barrier requirements.

The west wall of the SFP is designed as a safety-related structure to protect the spent fuel and contain borated water under specified conditions. The design forces of this shear wall include hydrostatic load and hydrodynamic loads induced by the water in the SFP and include thermal load due to the temperature transient in a loss-of-coolant accident (LOCA) event.

The east wall of the FHA is designed as a safety-related structure to protect the fuel handling system. As an exterior wall, the design forces of this shear wall meet the missile barrier requirements.

Design Summary

The structural design for the critical section provides reinforcement to resist the element forces and moments as described below:

a. North wall of north MSIV house: The north wall of the north MSIV house extends from the top of the basemat at El. 55 ft 0 in to the roof at El. 157 ft 0 in. It is 1.52 m (5 ft) thick. The compartment pressurization caused by a high-energy line break (HELB) is considered in the design of the MSIV house wall. The governing forces and moments from the analysis are summarized in Table 3.8A-28. The reinforcement arrangements of the north wall of the north MSIV house are shown in Table 3.8A-29, Figure 3.8A-36, and Figure 3.8A-37.

b. North wall of the north AFWST: The dimensions of the AFWST are 14.63 m (48 ft) × 7.92 m (26 ft). Tank bases are located at El. 100 ft 0 in. The maximum water level is 9.55 m (31 ft 4 in.) from the bottom of the AFWST. The wall thickness is 1.22 m (4 ft). The local structural analysis is additionally performed for local loads such as hydrostatic pressures loads acting on the walls of the north wall of the north AFWST. The governing forces and moments from the analysis are summarized in Table 3.8A-28. Reinforcement arrangement of the north wall of the north AFWST wall is shown in Table 3.8A-29 and Figure 3.8A-38.

c. West wall of the MCR: The wall at column line 12 is a shear wall on the west side of the MCR. It extends from the top of basemat at El. 55 ft 0 in to the top of the roof. The wall thickness is 0.92 m (3 ft) and 1.22 m (4 ft). The west wall of the
MCR is designed for the applicable loads including dead load, live load, hydrostatic load, static/dynamic lateral soil pressure loads, and seismic loads. The wall that is a part of the MCR boundary is from El. 156 ft 0 in to El. 174 ft 0 in. The governing forces and moments from the analysis are summarized in Table 3.8A-28. Reinforcement arrangements of the west wall of the MCR are shown in Table 3.8A-29, Figure 3.8A-39, and Figure 3.8A-40.

d. West wall of the SFP: The size of the west wall of the SFP is 12.80 m (42 ft) × 10.82 m (35.5 ft). Tank bases are located at El. 114 ft 0 in. The maximum water level is 12.24 m (40 ft 2 in.) from the bottom of the SFP. The wall thickness is 2.13 m (7 ft). The local structural analysis is performed for local loads such as the hydrostatic pressures loads and thermal loads. The governing forces and moments from the analysis are summarized in Table 3.8A-28. Reinforcement arrangement of the west wall of the SFP is shown in Table 3.8A-29 and Figure 3.8A-41.

e. East wall of the FHA: The FHA extends from the top of the concrete floor at El. 156 ft 0 in to the roof at El. 213 ft 6 in. The wall thickness is 1.22 m (4 ft). The FHA wall is designed for the applicable loads including dead load, live load, and seismic loads. The governing forces and moments from the analysis are summarized in Table 3.8A-28. Reinforcement arrangement of the fuel handling wall is shown in Table 3.8A-29, Figure 3.8A-42, and Figure 3.8A-43.

Table 3.8A-29 presents the margins of safety of reinforcement in the AB shear wall. The margin of safety is the ratio of required reinforcement and provided reinforcement.

3.8A.2.4.3 Concrete Slabs

Description

The concrete slabs of the AB are supported either by concrete walls or concrete beams. The thicknesses of the slabs are from 0.46 m (1.5 ft) to 1.22 m (4 ft) except for the FHA, which meets the radiation shielding requirement. Span lengths of one-way slabs in the AB do not exceed 7.32 m (24 ft).

Analysis and Design Methods

The concrete slabs are structurally analyzed using the FE analysis program GTSTRUDL. A separate analysis model simulating each floor level is prepared (Figure 3.8A-29) and
evaluated for each specified design load condition. To incorporate the proper vertical seismic load on each slab, the pseudo-static analysis is performed by using the response spectrum of corresponding floor.

The thickness of the slab is generally determined by the requirement of radiation and missile protection rather than that of structural integrity. Based on the enveloped results of FE analysis, the slab reinforcements are determined for flexure and out-of-plane shear in accordance with ACI 349. Additionally, the design of the slab at the connection of the shear wall considers the diaphragm shear (in-plane shear), which is transferred to the shear wall. The diaphragm shear considers the effects of lateral loads on the slab for the seismic loading cases, and the concrete slabs are designed for the diaphragm shear, which is determined by in-plane stress of the slab from the global structural static analysis. The reinforcement of the slab adjacent to the shear wall is determined for flexure and combined shear, including the diaphragm shear of the AB in accordance with ACI 349.

Selection of Critical Sections

The typical design for the slab of the auxiliary building is performed by the aforementioned procedure. Aside from that, the critical sections of slab meet the more stringent requirement to provide reasonable assurance of the function and safety of the system on the shear wall or consider potential load induced by abnormal and extreme environmental condition. The critical sections are selected using the following criteria:

a. The section contains the primary safety-related system

b. The section considers the specific load case under abnormal and extreme environmental conditions

The floor slab of the emergency diesel generator room at El. 100 ft 0 in is designed as a safety-related structure to sustain the emergency system including the emergency diesel generator under specified conditions. The design forces of that slab include the supporting load of equipment such as the emergency diesel generator.

The pool bottom slab of the spent fuel pool at El. 113 ft 0 in is designed as a safety-related structure to support and protect the spent fuel rack and to hold the massive borated water under specified conditions. The design forces of that slab include hydrostatic load and hydrodynamic load induced by water in the SFP and include thermal load due to temperature transient in the LOCA event.
The floor slab below the main steam enclosure at El. 137 ft 6 in is designed as a safety-related structure to provide reasonable assurance of safety of the internal system. The design forces of the slab include the compartment pressure due to a pipe break in the main steam line system.

**Design Summary**

The slab design results for the critical section are summarized as follows:

a. Concrete slab of the EDG room at El. 100 ft 0 in: The dimensions of the EDG room are 15.8 m (52 ft) × 25.2 m (82.5 ft), as shown in Figure 3.8A-45. The slab thicknesses of the emergency diesel generator room are 0.61 m (2 ft) and 1.22 m (4 ft). The EDG room is supported by the shear walls and intersected concrete beams. The FE analysis is carried out for the slabs located at El. 100 ft 0 in. The heavy equipment load of the EDG and the live load related the equipment removals are considered in the slab design. The governing forces and moments from the analysis are summarized in Table 3.8A-30, and the slab reinforcements considering the diaphragm shear are shown in Table 3.8A-33.

b. Concrete slab of the SFP at El. 114 ft 0 in: Concrete slab of the spent fuel pool is 12.8 m (42 ft) × 10.8 m (35.5 ft) with 1.85 m (6 ft 1 in.) thick. Hydrostatic pressure loads, hydrodynamic pressure loads, and thermal loads are considered in the SFP slab design. Thermal loads for the state of abnormal operation are considered to be 82.2 °C (180 °F) for inside and 21.1 °C (70 °F) for outside. The structural integrity of the slabs for thermal and mechanical loads is checked by the DARTEM program. The design forces and moments of the SFP slab are summarized in Table 3.8A-31, and the slab reinforcements are shown in Figure 3.8A-46.

c. Concrete slab of the main steam enclosure (MSE) at El. 137 ft 6 in: MSE slab measures 7.3 m (24 ft) wide by 37.3 m (122 ft 6 in.) long. The slab thickness is 1.22 m (4 ft). In case of the MSE slab, compartment pressure loads due to HELB are considered in the slab design. The design member forces and moments of the MSE slab are summarized in Table 3.8A-31, and the slab reinforcements considering diaphragm shear are shown in Table 3.8A-33.
Table 3.8A-33 presents the margins of safety of reinforcement at the critical section in the AB slab. The margin of safety is the ratio of required reinforcement and provided reinforcement.

3.8A.3 Emergency Diesel Generator Building

3.8A.3.1 Structural Description and Geometry

The EDG building block consists of two independent buildings, the EDG building at El. 100 ft 0 in and the diesel fuel oil tank (DFOT) building at El. 63 ft 0 in. The two basemats are separated by the isolation gap of 900 mm (3 ft). The EDG building houses two additional generators, and the DFOT building houses the DFOT.

The lateral load-resisting system of this building is composed of a diaphragm slab at roof level and shear walls monolithically interconnected with the roof. The vertical load-resisting system consists of four columns and the shear walls. The DFOT building is a typical box-type structure, which consists of the slabs, shear walls, and basemat. The building description is provided in Subsection 3.8A.4.1.2. The arrangements of the structure are shown in Figures 3.8A-63 and 3.8A-64.

3.8A.3.2 Structural Materials

The major materials of the EDG building block are concrete and reinforcing bars. The properties of the materials are identical to those of the AB, which is described in Subsection 3.8A.2.2.

3.8A.3.3 Loads and Load Combinations

3.8A.3.3.1 Design Loads

The following are the major design loads considered in the design of the EDG building.

Dead Load (D)

Dead load includes the weight of structures and the weight of permanently attached major equipment such as tanks, machinery, and cranes.

The minimum attachment loads include equipment loads lower than 10 kips such as piping, cable trays, and HVAC loads:
a. Concrete floors: 9.6 kN/m² (200 psf)
b. Roof floors: 7.2 kN/ m² (150 psf)
c. Interior wall: 1.0 kN/ m² (20 psf)
d. Exterior wall: 0.5 kN/ m² (10 psf)

Equipment loads heavier than or equal to 10 kips such as the DFOT, D/G stack, and diesel generator exhaust silencer are considered concentrated weight.

**Live Load (L)**

Live loads are conventional floor loads to account for occupancy, maintenance, equipment removal, equipment laydown, and other loads that vary in intensity. The design values are identical to those of the AB described in Subsection 3.8A.2.3.1.

**Wind Load (W)**

Wind load is the equivalent static load generated by the design wind velocity and is calculated in accordance with ASCE 7. The EDG building is designed for a 100-year recurrence interval wind.

**Safe Shutdown Earthquake Load (Eₛ)**

In the structural analysis for the EDG building, seismic loads are considered with the equivalent static load method involving equivalent horizontal and vertical static forces.

3.8A.3.3.2 **Load Combinations**

The load combinations are addressed in Subsection 3.8.4.3.6 and are used for analysis and design of the EDG building and associated components.

3.8A.3.4 **Analysis and Design for Critical Sections**

This section summarizes the analysis and design for critical sections of the EDG Building. The critical sections are listed below. The description of critical section, analysis and design methods, and design summary are provided for each critical section. The locations of critical sections are shown in Figure 3.8A-49.
3.8A.3.4.1 Basemat

Description

The EDG building and DFOT building are built on a separate concrete-reinforced mat foundation with a thickness of 1.21 m (4 ft). The two basemats are separated by an isolation gap of 900 mm (3 ft).

Vertical loads are transmitted through columns and walls down to the basemat. Lateral loads are transmitted to the basemat by a load-resisting system, such as shear walls and concrete slab. The foundation provides resistance to the transmitted loads through friction and bearing. Friction between the basemat and the foundation is to prevent sliding of the structure. Vertical load and load due to overturning of the structure are resisted by the foundation bearing.

Analysis and Design Methods

Structural analysis for the EDG building is performed using the ANSYS structural analysis program. Global structural analysis is first conducted to compute all member forces of shear wall in the EDG building. The computed member forces are used in the structural design. All fixed boundary conditions are imposed in the basemat and the dead, live, wind, seismic force, and soil pressure are applied.

The additional load is buoyant. This analysis used spring element to consider the influence of soil. All fixed boundary conditions are imposed at the end node of spring element. Figure 3.8A-30 shows the full FEM for the global structural analysis, and Figure 3.8A-31 shows the basemat model of the EDG building. The design forces and moments for the EDG building basemat are summarized in Table 3.8A-34.

The structural design for the EDG building basemat is performed in accordance with all requirements of ACI 349. Required reinforcements are calculated based on the governing required capacities obtained from FE analysis.

Design Summary

For the EDG building area, the maximum required moment capacities obtained from the basemat analysis are 947.3 k-ft/ft for the N-S direction and 794.8 k-ft/ft for the E-W direction. Upon the required design moment capacities, the EDG building basemat is
provided with 2-#11 bar at 305 mm (12 in.) spacing in each direction at each face and shear bar #5 at 305 mm (12 in.) spacing is provided to a 1.21 m (4 ft) thick area. Figure 3.8A-50 shows the typical sections of the EDG building basemat. The basemat reinforcement and margins of safety are shown in Table 3.8A-37. The margin of safety is the ratio of required reinforcement and provided requirement.

Stability Check

EDG and DFOT basemat structure is evaluated for stability against overturning, sliding, and flotation. The calculated factors of safety against overturning, sliding, and floatation for the load combinations meet the criteria of Section II of SRP 3.8.5, as shown in Table 3.8A-14.

The sliding and overturning factors of safety are determined using load combination containing dead load (D), wind load (W), SSE (E_s), and buoyant load at normal (H_e). The flotation factor of safety is determined based on dead load (D) and buoyant force at flood (H_s).

The normal design ground water elevation is EL. 96 ft 8 in. The extreme ground water elevation is the same as plant grade level (EL. 98 ft 8 in.) considering probable maximum flood.

In the earthquake load, axial force, shear force, and moment due to horizontal and vertical excitation of the structure are obtained from seismic analysis. Since seismic load governs over tornado load, the tornado load is not necessary to be considered in the stability evaluation. The wind load combination (D + H_e + W) in Table 3.8A-14 is considered for the EDG building though the seismic load governs over the wind load, because the allowable factor of safety for wind is 1.5, which is different from 1.1 for the seismic load. Because the DFOT building is a buried structure and the wind load is not applicable to the overturning and sliding check. A summary of overturning, sliding, and flotation check is shown provided in Table 3.8A-38.

Settlement Check

Differential settlements are divided by the differential settlement within the EDG building basemat and the differential settlement within DFOT building. For the differential settlements within the each basemat, the static (dead and live loads) loading case is calculated.
The distance of approximately 15.24 m (50 ft) is selected to check the differential settlement. Table 3.8A-39 shows the differential settlements of each soil profile. The maximum differential settlement for EDG building per 15.24 m (50 ft) is 4.623 mm (0.182 in.). The maximum differential settlement for DFOT building per 15.24 m (50 ft) is 9.754 mm (0.384 in.).

The differential settlement of each soil profiles between the NI common basemat and EDG building is checked. The maximum differential settlement between the NI common basemat and EDG building is 17.932 mm (0.706 in.).

The differential settlement of each soil profiles between the NI common basemat and DFOT building is checked. The maximum differential settlement between the NI common basemat and DFOT building is 46.660mm (1.837 in.).

The differential settlement of each soil profiles between the EDG building and DFOT building is checked. The maximum differential settlement between the EDG building and DFOT building is 23.495 mm (0.925 in.).

Figure 3.8A-54 and Figure 3.8A-55 show the node locations at the bottom of the EDG & DFOT basemat for checking the settlements. The analysis of multiple of settlements (long and short term) will use these nodes.

Bearing Pressure Check

The bearing pressures of the EDG building basemat and DFOT basemat are evaluated for soil profile S (weakest), S04 (moderate), S08 (strongest) under static and dynamic loading conditions. The analysis and design methods used for bearing pressure evaluation is identical as that used for the design described in 3.8A.3.4.1.

The static and dynamic bearing pressure is determined as the soil spring forces divided by the tributary area of the soil spring under static (dead and live load) and dynamic conditions. The maximum static bearing pressure of 396.0kPa (8,270 lb/ft2) in the EDGB basemat is obtained from the basemat analysis results of soil profile S08. The maximum static bearing pressure of 352.9kPa (7,370 lb/ft2) in the DFOT basemat is obtained from the basemat analysis results of soil profile S08. The maximum dynamic bearing pressure of 861.37kPa (17,990 lb/ft2) in the EDGB basemat is obtained from the basemat analysis results of soil profile S08. The maximum dynamic bearing pressure of 866.7kPa (18,100 lb/ft2) in the DFOT basemat is obtained from the basemat analysis results of soil profile S08.
3.8A.3.4.2 **Shear Walls**

**Description**

The shear walls and slabs of the EDG building representing the primary lateral load-resisting system are designed against seismic or extreme wind-related loads. The concrete slab distributes lateral forces through diaphragm action to the shear walls as in-plane loads in proportion to the relative stiffness of the shear walls. These in-plane shear forces are transferred down to the basemat foundation as in-plane shear forces and moments of the shear walls.

The in-plane shear forces and moments, which are obtained from the seismic analysis results, are combined with the applicable out-of-plane loads to determine the quantity and distribution of vertical and horizontal reinforcing steel of the EDG building shear walls.

**Analysis and Design Methods**

The global static analysis is performed using the ANSYS structural analysis program to evaluate element forces of the shear walls. Shear walls are modeled by the Shell 181 element in ANSYS, while Beam 188 is used to model the concrete frame. The Link 180 element is applied to consider the effects of soil spring on the basemat analysis. The FEM representing the whole EDG building for structural analysis is shown in Figure 3.8A-30.

Before determining the required reinforcing steel in the shear walls, the acceptability of concrete shear stresses is checked. In-plane shear stress, out-of-plane shear stress, and combined concrete shear stresses are calculated for each shear wall to provide reasonable assurance that the shear stress does not exceed the acceptable values in accordance with ACI 349. The horizontal and vertical reinforcements required for the shear walls are computed by combining the individual reinforcement requirements resulting from the various load conditions.

The exterior walls below the grade are designed to resist the worst-case lateral earth pressure loads (static and dynamic), soil surcharge loads, and loads due to groundwater (static and dynamic). Lateral earth pressure is equal to the summation of the static earth pressure plus the dynamic earth pressure. The dynamic earth pressure are determined as the governing case between the dynamic earth pressures calculated in accordance with ASCE 4-98 (Reference 47 in Subsection 3.8.7) and those calculated based on the results of
the SSI/SSSI analyses. The hydrodynamic effect of pure water is determined based on the hydrodynamic formula suggested by Matuo and O’Hara (Reference 48 in Subsection 3.8.7).

Design Summary

Table 3.8A-35 shows the summary of the element forces for the critical shear walls of the EDG building. The final vertical and horizontal reinforcing steel required in each shear wall are determined based on both calculation requirements and practicable considerations such as placement of the reinforcing steel in the shear walls. The results of the shear wall analysis and design, in the form of horizontal and vertical reinforcement quantities required and provided for each shear wall at basemat level and margins of safety, are summarized in Table 3.8A-36. The margin of safety is the ratio of required reinforcement and provided reinforcement.

The structural design for the critical section provides reinforcement to resist element forces and moments as described as follows:

a. Center wall located in the middle of the EDG building: The center wall of the EDG building extends from the top of the basemat at El. 100 ft 6 in to the roof at El. 135 ft 0 in. It is 0.76 m (2 ft 6 in.) thick. The reinforcing steel arrangements of the center wall of the EDG building are shown in Figure 3.8A-51.

b. West wall of the EDG building: The west wall of the EDG building extends from the top of the basemat at El. 100 ft 0 in to the roof at El. 135 ft 0 in. It is 0.9 m (3 ft) thick. The reinforcing steel arrangements of the west wall of the EDG building are shown in Figure 3.8A-52.
### Table 3.8A-1

**Allowable Stresses for Containment Materials**

<table>
<thead>
<tr>
<th>Material and Stress Type</th>
<th>Service Loads</th>
<th>Factored Loads</th>
</tr>
</thead>
<tbody>
<tr>
<td>Concrete</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( P_m )</td>
<td>0.30 ( f'_c )</td>
<td>0.60 ( f'_c )</td>
</tr>
<tr>
<td>( P_m + b )</td>
<td>0.45 ( f'_c )</td>
<td>0.75 ( f'_c )</td>
</tr>
<tr>
<td>( (P + Q)_m )</td>
<td>0.45 ( f'_c )</td>
<td>0.75 ( f'_c )</td>
</tr>
<tr>
<td>( (P + Q)_m + b )</td>
<td>0.60 ( f'_c )</td>
<td>0.85 ( f'_c )</td>
</tr>
<tr>
<td>Reinforcing Steel</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Compression</td>
<td>0.50 ( f_y )</td>
<td>0.90 ( f_y )</td>
</tr>
<tr>
<td>Tension</td>
<td>0.50 ( f_y )</td>
<td>0.90 ( f_y )</td>
</tr>
</tbody>
</table>

\( P_m \) = Primary membrane stresses  
\( P_m + b \) = Primary membrane plus bending stresses  
\( (P + Q)_m \) = Primary plus secondary membrane stresses  
\( (P + Q)_m + b \) = Primary plus secondary membrane plus bending stresses  
\( f'_c \) = Minimum specified compressive strength of concrete  
\( f_y \) = Minimum specified yield strength of reinforcing steel
Table 3.8A-2 (1 of 2)

Section Forces of Containment Wall and Dome Design Sections

### Wall-Basemat Junction Area

<table>
<thead>
<tr>
<th>N_φ (kip/ft)</th>
<th>M_φ (kip-ft/ft)</th>
<th>Q_{Rφ} (kip/ft)</th>
<th>N_θ (kip/ft)</th>
<th>M_θ (kip-ft/ft)</th>
<th>Q_{Rθ} (kip/ft)</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>376.56</td>
<td>-422.17</td>
<td>110.59</td>
<td>268.29</td>
<td>-78.46</td>
<td>8.05</td>
<td>Meridional Inside</td>
</tr>
<tr>
<td>363.30</td>
<td>253.54</td>
<td>38.88</td>
<td>333.86</td>
<td>54.11</td>
<td>5.66</td>
<td>Meridional Outside</td>
</tr>
<tr>
<td>331.52</td>
<td>52.21</td>
<td>6.19</td>
<td>373.59</td>
<td>-7.74</td>
<td>0.38</td>
<td>Hoop Inside</td>
</tr>
<tr>
<td>328.69</td>
<td>222.00</td>
<td>5.12</td>
<td>368.07</td>
<td>80.67</td>
<td>3.52</td>
<td>Hoop Outside</td>
</tr>
</tbody>
</table>

### Mid-Height Level of Wall

<table>
<thead>
<tr>
<th>N_φ (kip/ft)</th>
<th>M_φ (kip-ft/ft)</th>
<th>Q_{Rφ} (kip/ft)</th>
<th>N_θ (kip/ft)</th>
<th>M_θ (kip-ft/ft)</th>
<th>Q_{Rθ} (kip/ft)</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>198.93</td>
<td>-271.97</td>
<td>27.44</td>
<td>241.72</td>
<td>29.67</td>
<td>11.22</td>
<td>Meridional Inside</td>
</tr>
<tr>
<td>233.67</td>
<td>124.91</td>
<td>11.05</td>
<td>288.59</td>
<td>152.77</td>
<td>4.50</td>
<td>Meridional Outside</td>
</tr>
<tr>
<td>-116.11</td>
<td>-53.26</td>
<td>3.34</td>
<td>373.32</td>
<td>-192.54</td>
<td>-38.69</td>
<td>Hoop Inside</td>
</tr>
<tr>
<td>-65.44</td>
<td>55.70</td>
<td>0.04</td>
<td>334.88</td>
<td>179.59</td>
<td>-1.67</td>
<td>Hoop Outside</td>
</tr>
</tbody>
</table>

### Polar Crane Bracket Level and Springline

<table>
<thead>
<tr>
<th>N_φ (kip/ft)</th>
<th>M_φ (kip-ft/ft)</th>
<th>Q_{Rφ} (kip/ft)</th>
<th>N_θ (kip/ft)</th>
<th>M_θ (kip-ft/ft)</th>
<th>Q_{Rθ} (kip/ft)</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>41.49</td>
<td>-236.69</td>
<td>-4.45</td>
<td>152.44</td>
<td>-112.27</td>
<td>26.10</td>
<td>Meridional Inside</td>
</tr>
<tr>
<td>76.39</td>
<td>85.49</td>
<td>12.72</td>
<td>170.73</td>
<td>6.75</td>
<td>25.74</td>
<td>Meridional Outside</td>
</tr>
<tr>
<td>-8.07</td>
<td>-100.63</td>
<td>-2.30</td>
<td>389.97</td>
<td>-286.61</td>
<td>11.49</td>
<td>Hoop Inside</td>
</tr>
<tr>
<td>10.98</td>
<td>-78.75</td>
<td>-7.73</td>
<td>393.01</td>
<td>-274.63</td>
<td>-8.64</td>
<td>Hoop Outside</td>
</tr>
</tbody>
</table>
### Equipment Hatch

<table>
<thead>
<tr>
<th>$N_\phi$ (kip/ft)</th>
<th>$M_\phi$ (kip-ft/ft)</th>
<th>$Q_{R\phi}$ (kip/ft)</th>
<th>$N_\theta$ (kip/ft)</th>
<th>$M_\theta$ (kip-ft/ft)</th>
<th>$Q_{R\theta}$ (kip/ft)</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>633.51</td>
<td>-900.58</td>
<td>59.89</td>
<td>443.77</td>
<td>-582.97</td>
<td>107.87</td>
<td>Meridional Inside</td>
</tr>
<tr>
<td>633.51</td>
<td>1,391.59</td>
<td>59.89</td>
<td>443.77</td>
<td>1,347.08</td>
<td>107.87</td>
<td>Meridional Outside</td>
</tr>
<tr>
<td>327.89</td>
<td>-375.99</td>
<td>-62.95</td>
<td>984.87</td>
<td>-1,121.67</td>
<td>-39.04</td>
<td>Hoop Inside</td>
</tr>
<tr>
<td>400.38</td>
<td>1,127.27</td>
<td>-14.59</td>
<td>682.06</td>
<td>1,119.63</td>
<td>-10.57</td>
<td>Hoop Outside</td>
</tr>
</tbody>
</table>

### Personnel Airlock

<table>
<thead>
<tr>
<th>$N_\phi$ (kip/ft)</th>
<th>$M_\phi$ (kip-ft/ft)</th>
<th>$Q_{R\phi}$ (kip/ft)</th>
<th>$N_\theta$ (kip/ft)</th>
<th>$M_\theta$ (kip-ft/ft)</th>
<th>$Q_{R\theta}$ (kip/ft)</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>742.23</td>
<td>-641.62</td>
<td>47.73</td>
<td>821.82</td>
<td>-538.01</td>
<td>17.70</td>
<td>Meridional Inside</td>
</tr>
<tr>
<td>596.38</td>
<td>1,586.38</td>
<td>66.48</td>
<td>830.83</td>
<td>687.51</td>
<td>38.89</td>
<td>Meridional Outside</td>
</tr>
<tr>
<td>595.99</td>
<td>-678.43</td>
<td>58.22</td>
<td>855.75</td>
<td>-615.76</td>
<td>-6.09</td>
<td>Hoop Inside</td>
</tr>
<tr>
<td>595.99</td>
<td>1,558.64</td>
<td>58.22</td>
<td>855.75</td>
<td>621.13</td>
<td>-6.09</td>
<td>Hoop Outside</td>
</tr>
</tbody>
</table>

### Dome

<table>
<thead>
<tr>
<th>$N_\phi$ (kip/ft)</th>
<th>$M_\phi$ (kip-ft/ft)</th>
<th>$Q_{R\phi}$ (kip/ft)</th>
<th>$N_\theta$ (kip/ft)</th>
<th>$M_\theta$ (kip-ft/ft)</th>
<th>$Q_{R\theta}$ (kip/ft)</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>84.76</td>
<td>67.80</td>
<td>-3.09</td>
<td>244.55</td>
<td>73.63</td>
<td>-0.37</td>
<td>Meridional Inside</td>
</tr>
<tr>
<td>84.76</td>
<td>67.80</td>
<td>-3.09</td>
<td>244.55</td>
<td>73.63</td>
<td>-0.37</td>
<td>Meridional Outside</td>
</tr>
<tr>
<td>84.76</td>
<td>67.80</td>
<td>-3.09</td>
<td>244.55</td>
<td>73.63</td>
<td>-0.37</td>
<td>Hoop Inside</td>
</tr>
<tr>
<td>84.76</td>
<td>67.80</td>
<td>-3.09</td>
<td>244.55</td>
<td>73.63</td>
<td>-0.37</td>
<td>Hoop Outside</td>
</tr>
</tbody>
</table>

$N_\phi$ = Meridional Force with Tangential Shear Force  
$M_\phi$ = Meridional Moment with Torsional Moment  
$QR_\phi$ = Meridional Radial Shear Force  
$N_\theta$ = Hoop Force with Tangential Shear Force  
$M_\theta$ = Hoop Moment with Torsional Moment  
$QR_\theta$ = Hoop Radial Shear Force
Reinforcing Details of Containment Wall and Dome Design Sections

### Wall-Basemat Junction Area

<table>
<thead>
<tr>
<th>Direction</th>
<th>Rebar Arrangement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Meridional</td>
<td>Layer 1: #18+#14 @ 0.85°</td>
</tr>
<tr>
<td></td>
<td>Layer 2: #18 @ 0.85°</td>
</tr>
<tr>
<td></td>
<td>Layer 1: #18+#18 @ 0.85°</td>
</tr>
<tr>
<td>Outside</td>
<td>Layer 1: #18 @ 12 in.</td>
</tr>
<tr>
<td></td>
<td>Layer 2: #11 @ 12 in.</td>
</tr>
<tr>
<td>Hoop</td>
<td>Layer 1: #18 @ 12 in.</td>
</tr>
<tr>
<td></td>
<td>Layer 2: #14 @ 12 in.</td>
</tr>
</tbody>
</table>

### Mid-Height Level of Wall

<table>
<thead>
<tr>
<th>Direction</th>
<th>Rebar Arrangement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Meridional</td>
<td>Layer 1: #18 @ 0.85°</td>
</tr>
<tr>
<td></td>
<td>Layer 1: #18+#14 @ 0.85°</td>
</tr>
<tr>
<td>Outside</td>
<td>Layer 1: #18+#11 @ 12 in.</td>
</tr>
<tr>
<td>Hoop</td>
<td>Layer 1: #18 @ 12 in.</td>
</tr>
<tr>
<td></td>
<td>Layer 2: #18 @ 12 in.</td>
</tr>
</tbody>
</table>

### Polar Crane Bracket Level and Springline

<table>
<thead>
<tr>
<th>Direction</th>
<th>Rebar Arrangement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Meridional</td>
<td>Layer 1: #14 @ 0.85°</td>
</tr>
<tr>
<td></td>
<td>Layer 2: #14 @ 0.85°</td>
</tr>
<tr>
<td>Outside</td>
<td>Layer 1: #18 @ 0.85°</td>
</tr>
<tr>
<td>Hoop</td>
<td>Layer 1: #18 + #14 @ 12 in.</td>
</tr>
<tr>
<td></td>
<td>Layer 1: #18 @ 12 in.</td>
</tr>
<tr>
<td></td>
<td>Layer 2: #18 @ 12 in.</td>
</tr>
</tbody>
</table>
### Equipment Hatch

<table>
<thead>
<tr>
<th>Direction</th>
<th>Inside</th>
<th>Rebar Arrangement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Meridional</td>
<td>Layer 1: #18+#14 @ 0.85° &lt;br&gt; Layer 2: #18 @ 0.85°</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Outside</td>
<td>Layer 1: #18+#18 @ 0.85° &lt;br&gt; Layer 2: #18+#18 @ 0.85° &lt;br&gt; Layer 3: #18+#18 @ 0.85°</td>
</tr>
<tr>
<td>Hoop</td>
<td>Inside</td>
<td>Layer 1: #18 @ 6 in. &lt;br&gt; Layer 2: #18 @ 6 in.</td>
</tr>
<tr>
<td></td>
<td>Outside</td>
<td>Layer 1: #18 @ 6 in. &lt;br&gt; Layer 2: #18 @ 12 in. &lt;br&gt; Layer 3: #18 @ 12 in. &lt;br&gt; Layer 4: #18 @ 12 in.</td>
</tr>
</tbody>
</table>

| Personnel Airlock |
|-------------------|---|---|
| Direction | Inside | Rebar Arrangement |
| Meridional | Layer 1: #18+#14 @ 0.85° <br> Layer 2: #18 @ 0.85° <br> Layer 3: #18 @ 0.85° |
|           | Outside | Layer 1: #18+#18 @ 0.8° <br> Layer 2: #18+#18 @ 0.8° <br> Layer 3: #18+#18 @ 0.85° |
| Hoop       | Inside | Layer 1: #18 @ 12 in. <br> Layer 2: #11 @ 12 in. |
|           | Outside | Layer 1: #18 @ 6 in. <br> Layer 2: #18 @ 12 in. <br> Layer 3: #18 @ 12 in. <br> Layer 4: #14 @ 12 in. |

### Dome

<table>
<thead>
<tr>
<th>Direction</th>
<th>Inside</th>
<th>Rebar Arrangement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Meridional</td>
<td>Layer 1: #14 @ 0.85°</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Outside</td>
<td>Layer 1: #18 @ 0.85°</td>
</tr>
<tr>
<td>Hoop</td>
<td>Inside</td>
<td>Layer 1: #18 @ 0.75°</td>
</tr>
<tr>
<td></td>
<td>Outside</td>
<td>Layer 1: #18+#11 @ 0.75°</td>
</tr>
</tbody>
</table>
### Wall-Basemat Junction Area (Rebar)

<table>
<thead>
<tr>
<th></th>
<th>Meridional</th>
<th>Hoop</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mech.  + Thermal</td>
<td>Mechanical</td>
</tr>
<tr>
<td>Inside</td>
<td>(ksi)</td>
<td>Inside</td>
</tr>
<tr>
<td>Outside</td>
<td>(ksi)</td>
<td>Outside</td>
</tr>
<tr>
<td></td>
<td>41.7</td>
<td>43.2</td>
</tr>
<tr>
<td>Ratio (1)</td>
<td>1.29</td>
<td>1.25</td>
</tr>
</tbody>
</table>

### Mid-Height Level of Wall (Rebar)

<table>
<thead>
<tr>
<th></th>
<th>Meridional</th>
<th>Hoop</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mech.  + Thermal</td>
<td>Mechanical</td>
</tr>
<tr>
<td>Inside</td>
<td>(ksi)</td>
<td>Inside</td>
</tr>
<tr>
<td>Outside</td>
<td>(ksi)</td>
<td>Outside</td>
</tr>
<tr>
<td></td>
<td>46.8</td>
<td>38.4</td>
</tr>
<tr>
<td>Ratio (1)</td>
<td>1.15</td>
<td>1.41</td>
</tr>
</tbody>
</table>

### Polar Crane Bracket Level and Springline (Rebar)

<table>
<thead>
<tr>
<th></th>
<th>Meridional</th>
<th>Hoop</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mech.  + Thermal</td>
<td>Mechanical</td>
</tr>
<tr>
<td>Inside</td>
<td>(ksi)</td>
<td>Inside</td>
</tr>
<tr>
<td>Outside</td>
<td>(ksi)</td>
<td>Outside</td>
</tr>
<tr>
<td></td>
<td>27.2</td>
<td>16.4</td>
</tr>
<tr>
<td>Ratio (1)</td>
<td>1.99</td>
<td>3.29</td>
</tr>
</tbody>
</table>
### Equipment Hatch (Rebar)

<table>
<thead>
<tr>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
</tr>
</thead>
<tbody>
<tr>
<td>39.4</td>
<td>38.9</td>
<td>39.0</td>
<td>54.0</td>
<td>40.2</td>
<td>32.9</td>
<td>37.9</td>
<td>48.9</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Ratio (1)</th>
<th>Ratio (1)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.37</td>
<td>1.39</td>
</tr>
</tbody>
</table>

(1) Ratio = allowable stress (0.9F_y) / actual stress

(2) The reinforcement meets the requirement of CC-3422.1 of the ASME Code.

### Personnel Airlock (Rebar)

<table>
<thead>
<tr>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
</tr>
</thead>
<tbody>
<tr>
<td>31.7</td>
<td>37.9</td>
<td>28.7</td>
<td>54.0</td>
<td>51.3</td>
<td>45.6</td>
<td>43.5</td>
<td>53.0</td>
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<table>
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### Dome (Rebar)

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<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
<th>Inside (ksi)</th>
<th>Outside (ksi)</th>
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<td>50.1</td>
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<td>50.3</td>
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<table>
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<tbody>
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</table>

(1) Ratio = allowable stress (0.9F_y) / actual stress

(2) The reinforcement meets the requirement of CC-3422.1 of the ASME Code.
### Wall-Basemat Junction Area (Concrete)

<table>
<thead>
<tr>
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<tbody>
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<td>Mech. + Thermal</td>
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<td>-2.1</td>
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<tr>
<td>Outside (ksi)</td>
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<tr>
<td><strong>Ratio (1)</strong></td>
<td><strong>Ratio (2)</strong></td>
<td><strong>Ratio (1)</strong></td>
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<td>1.77</td>
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<td>-1.9</td>
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<tr>
<td>Outside (ksi)</td>
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<td></td>
</tr>
<tr>
<td><strong>Ratio (3)</strong></td>
<td><strong>Ratio (4)</strong></td>
<td><strong>Ratio (3)</strong></td>
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<tr>
<td>1.86</td>
<td>1.86</td>
<td>1.76</td>
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### Mid-Height Level of Wall (Concrete)

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</tr>
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<td><strong>Ratio (2)</strong></td>
<td><strong>Ratio (1)</strong></td>
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<tr>
<td>Outside (ksi)</td>
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<tr>
<td><strong>Ratio (3)</strong></td>
<td><strong>Ratio (4)</strong></td>
<td><strong>Ratio (3)</strong></td>
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<tr>
<td>2.10</td>
<td>2.10</td>
<td>1.77</td>
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### Polar Crane Bracket Level and Springline (Concrete)

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<td>Inside (ksi)</td>
</tr>
<tr>
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<tr>
<td><strong>Ratio (1)</strong></td>
<td><strong>Ratio (2)</strong></td>
<td><strong>Ratio (1)</strong></td>
</tr>
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<td>Inside (ksi)</td>
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<td>-2.4</td>
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<td><strong>Ratio (3)</strong></td>
<td><strong>Ratio (4)</strong></td>
<td><strong>Ratio (3)</strong></td>
</tr>
<tr>
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<td>2.71</td>
<td>1.86</td>
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</table>

### Equipment Hatch (Concrete)

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<th>Hoop</th>
</tr>
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<tbody>
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<td>Mech. + Thermal</td>
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<td>Outside (ksi)</td>
<td>Inside (ksi)</td>
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<tr>
<td>-2.3</td>
<td>-1.7</td>
<td>-2.6</td>
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<tr>
<td><strong>Ratio (1)</strong></td>
<td><strong>Ratio (2)</strong></td>
<td><strong>Ratio (1)</strong></td>
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<td>Inside (ksi)</td>
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<tr>
<td>-1.9</td>
<td>-1.7</td>
<td>-2.6</td>
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<tr>
<td><strong>Ratio (3)</strong></td>
<td><strong>Ratio (4)</strong></td>
<td><strong>Ratio (3)</strong></td>
</tr>
<tr>
<td>1.92</td>
<td>2.15</td>
<td>1.76</td>
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### Personnel Airlock (Concrete)

<table>
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<th>Meridional</th>
<th>Hoop</th>
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<td>Inside (ksi)</td>
<td>Outside (ksi)</td>
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<tr>
<td>Membrane</td>
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<tr>
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<td>-1.4</td>
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<tr>
<td>Ratio (3)</td>
<td>2.23</td>
<td>2.34</td>
<td>1.82</td>
<td>3.30</td>
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</tbody>
</table>

(1) Ratio = allowable compression stress for mechanical load (0.75 f'_c) / actual stress
(2) Ratio = allowable compression stress for mechanical plus thermal load (0.85 f'_c) / actual stress
(3) Ratio = allowable compression stress for mechanical load (0.6 f'_c) / actual stress
(4) Ratio = allowable compression stress for mechanical plus thermal load (0.75 f'_c) / actual stress

### Dome (Concrete)

<table>
<thead>
<tr>
<th>Membrane + Bending</th>
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<th>Hoop</th>
<th>Meridional</th>
<th>Hoop</th>
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<tbody>
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<td>Inside (ksi)</td>
<td>Outside (ksi)</td>
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<tr>
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<td>-1.1</td>
<td>-2.3</td>
<td>-1.1</td>
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<tr>
<td>Ratio (3)</td>
<td>3.14</td>
<td>3.14</td>
<td>1.93</td>
<td>4.09</td>
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</tbody>
</table>

(1) Ratio = allowable compression stress for mechanical load (0.75 f'_c) / actual stress
(2) Ratio = allowable compression stress for mechanical plus thermal load (0.85 f'_c) / actual stress
(3) Ratio = allowable compression stress for mechanical load (0.6 f'_c) / actual stress
(4) Ratio = allowable compression stress for mechanical plus thermal load (0.75 f'_c) / actual stress
## Critical Sections of RCB Basemat

<table>
<thead>
<tr>
<th>Design Section</th>
<th>Location</th>
<th>Height (ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Section-01</td>
<td>Tendon Gallery Outside Area</td>
<td>33</td>
</tr>
<tr>
<td>Section-02</td>
<td>Tendon Gallery Upper Area</td>
<td>10</td>
</tr>
<tr>
<td>Section-03</td>
<td>Tendon Gallery Below Area</td>
<td>10</td>
</tr>
</tbody>
</table>
| Section-04     | R = 63.25’ to 70.5’  
                | at El. 45’ to El. 78’                             | 33          |
| Section-05     | R = 42.5’ to 63.25’  
                | Exclude Design Section-06, -07, -08               | 33          |
| Section-06     | Cavity Area                                        | 11          |
|                | at El. 55’ to 66’                                  |             |
| Section-07     | x = 22’ to 39’ and y = 0’ to 18.75’               | 21          |
|                | at El. 55’ to 76’                                  |             |
| Section-08     | x, y = 0’ to 42.5’  
                | Exclude Design Section -06, -07                   | 23          |
### Table 3.8A-6

**Flexural Forces of RCB Basemat (Service load)**

<table>
<thead>
<tr>
<th>Design Section</th>
<th>E-W or Radial Direction</th>
<th>N-S or Hoop Direction</th>
</tr>
</thead>
<tbody>
<tr>
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<td>F_x (kips/ft)</td>
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<tr>
<td>Section-01</td>
<td>411.98</td>
<td>-689.04</td>
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<tr>
<td>Section-02</td>
<td>1,202.21</td>
<td>20.30</td>
</tr>
<tr>
<td>Section-03</td>
<td>1,876.71</td>
<td>276.72</td>
</tr>
<tr>
<td>Section-04</td>
<td>5,601.79</td>
<td>189.17</td>
</tr>
<tr>
<td>Section-05</td>
<td>7,300.82</td>
<td>431.25</td>
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<td>321.98</td>
<td>236.43</td>
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<td>2,093.17</td>
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<td>+M_y (kips-ft/ft)</td>
<td>F_y (kips/ft)</td>
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<td>216.13</td>
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<td>670.62</td>
<td>127.57</td>
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<tr>
<td>Section-08</td>
<td>2,685.55</td>
<td>203.26</td>
</tr>
</tbody>
</table>

Where,

+M_x: Maximum bending moment per unit width about local 2-axis (E-W or Radial)

F_x: Direct axial force per unit width in local 1-direction (E-W or Radial) when +M_x occurs

-M_x: Minimum bending moment per unit width about local 2-axis (E-W or Radial)

F_x: Direct axial force per unit width in local 1-direction (E-W or Radial) when -M_x occurs

+M_y: Maximum bending moment per unit width about local 1-axis (N-S or Hoop)

F_y: Direct axial force per unit width in local 1-direction (E-W or Radial) when +M_y occurs

-M_y: Minimum bending moment per unit width about local 1-axis (N-S or Hoop)

F_y: Direct axial force per unit width in local 2-direction (N-S or Hoop) when -M_y occurs
# APR1400 DCD TIER 2

## Table 3.8A-7

Flexural Forces of RCB Basemat (Factored load)

<table>
<thead>
<tr>
<th>Design Section</th>
<th>N-S or Hoop Direction</th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>+Mx (kips-ft/ft)</td>
<td>Fx (kips/ft)</td>
<td>-Mx (kips-ft/ft)</td>
<td>Fx (kips/ft)</td>
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<td>----------------</td>
<td>-----------------------</td>
<td>---</td>
<td>---</td>
<td>---</td>
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<td>-223.03</td>
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<td>312.49</td>
<td>-752.63</td>
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<td>980.91</td>
<td>365.09</td>
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<td>18.18</td>
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<table>
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<td>Fx (kips/ft)</td>
<td>-Mx (kips-ft/ft)</td>
<td>Fx (kips/ft)</td>
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<td>521.66</td>
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Where,

+Mx: Maximum bending moment per unit width about local 2-axis (E-W or Radial)

Fx: Direct axial force per unit width in local 1-direction (E-W or Radial) when +Mx occurs

-Mx: Minimum bending moment per unit width about local 2-axis (E-W or Radial)

Fx: Direct axial force per unit width in local 1-direction (E-W or Radial) when -Mx occurs

+My: Maximum bending moment per unit width about local 1-axis (N-S or Hoop)

Fy: Direct axial force per unit width in local 2-direction (N-S or Hoop) when +My occurs

-My: Minimum bending moment per unit width about local 1-axis (N-S or Hoop)

Fy: Direct axial force per unit width in local 2-direction (N-S or Hoop) when -My occurs
Table 3.8A-8

Shear Forces of RCB Basemat (Service load)

<table>
<thead>
<tr>
<th>Design Section</th>
<th>N_{xx} (kips/ft)</th>
<th>N_{yy} (kips/ft)</th>
<th>Q_{xz} (kips/ft)</th>
<th>Q_{yz} (kips/ft)</th>
<th>V_{max} (kips/ft)</th>
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<tr>
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<td>274.36</td>
<td>-203.02</td>
<td>-49.17</td>
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<td>-208.75</td>
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<td>-178.73</td>
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<td>166.19</td>
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<td>112.56</td>
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<td>39.00</td>
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<td>111.55</td>
<td>50.75</td>
<td>-249.73</td>
<td>335.54</td>
</tr>
</tbody>
</table>

Where,

- $Q_{xz}$: Transverse shear force per unit width in local 1-direction
- $Q_{yz}$: Transverse shear force per unit width in local 2-direction
- $N_{xx}$: Direct axial force per unit width in local 1-direction when $Q_{xz}$ and $Q_{yz}$ occurs
- $N_{yy}$: Direct axial force per unit width in local 2-direction when $Q_{xz}$ and $Q_{yz}$ occurs
- $V_{max}$: Design shear force per unit width
Table 3.8A-9

Shear Forces of RCB Basemat (Factored load)

<table>
<thead>
<tr>
<th>Design Section</th>
<th>( N_{xx} ) (kips/ft)</th>
<th>( N_{yy} ) (kips/ft)</th>
<th>( Q_{xz} ) (kips/ft)</th>
<th>( Q_{yz} ) (kips/ft)</th>
<th>( V_{\text{max}} ) (kips/ft)</th>
</tr>
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<tbody>
<tr>
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<td>-284.95</td>
<td>-149.30</td>
<td>-161.19</td>
<td>330.75</td>
</tr>
<tr>
<td>Section-04</td>
<td>-767.30</td>
<td>328.42</td>
<td>-185.84</td>
<td>-560.74</td>
<td>746.58</td>
</tr>
<tr>
<td>Section-05</td>
<td>-82.70</td>
<td>657.22</td>
<td>231.95</td>
<td>-727.95</td>
<td>959.91</td>
</tr>
<tr>
<td>Section-06</td>
<td>968.25</td>
<td>231.73</td>
<td>-125.58</td>
<td>363.97</td>
<td>527.52</td>
</tr>
<tr>
<td>Section-07</td>
<td>979.48</td>
<td>105.50</td>
<td>-273.05</td>
<td>-500.34</td>
<td>773.40</td>
</tr>
<tr>
<td>Section-08</td>
<td>981.22</td>
<td>262.71</td>
<td>515.15</td>
<td>-616.84</td>
<td>1,131.99</td>
</tr>
</tbody>
</table>

Where,
- \( Q_{xz} \): Transverse shear force per unit width in local 1-direction
- \( Q_{yz} \): Transverse shear force per unit width in local 2-direction
- \( N_{xx} \): Direct axial force per unit width in local 1-direction when \( Q_{xz} \) and \( Q_{yz} \) occurs
- \( N_{yy} \): Direct axial force per unit width in local 2-direction when \( Q_{xz} \) and \( Q_{yz} \) occurs
- \( V_{\text{max}} \): Design shear force per unit width
### Table 3.8A-10

**Flexural Rebar and Concrete Stresses and Margins of Safety for RCB Basemat**

#### Rebar

<table>
<thead>
<tr>
<th>Design Section</th>
<th>Service Load Combination</th>
<th>Factored Load Combination</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Allowable Stress (ksi)</td>
<td>Maximum Stress (ksi)</td>
</tr>
<tr>
<td>Section-01</td>
<td>30.00</td>
<td>27.13</td>
</tr>
<tr>
<td>Section-02</td>
<td>24.06</td>
<td>1.25</td>
</tr>
<tr>
<td>Section-03</td>
<td>12.46</td>
<td>2.41</td>
</tr>
<tr>
<td>Section-04</td>
<td>10.51</td>
<td>2.85</td>
</tr>
<tr>
<td>Section-05</td>
<td>14.08</td>
<td>2.13</td>
</tr>
<tr>
<td>Section-06</td>
<td>7.47</td>
<td>4.02</td>
</tr>
<tr>
<td>Section-07</td>
<td>7.25</td>
<td>4.14</td>
</tr>
<tr>
<td>Section-08</td>
<td>10.94</td>
<td>2.74</td>
</tr>
</tbody>
</table>

#### Concrete

<table>
<thead>
<tr>
<th>Design Section</th>
<th>Service Load Combination</th>
<th>Factored Load Combination</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Allowable Stress (ksi)</td>
<td>Maximum Stress (ksi)</td>
</tr>
<tr>
<td>Section-01</td>
<td>2.25</td>
<td>1.527</td>
</tr>
<tr>
<td>Section-02</td>
<td>0.782</td>
<td>2.88</td>
</tr>
<tr>
<td>Section-03</td>
<td>0.677</td>
<td>3.32</td>
</tr>
<tr>
<td>Section-04</td>
<td>0.261</td>
<td>8.62</td>
</tr>
<tr>
<td>Section-05</td>
<td>0.338</td>
<td>6.66</td>
</tr>
<tr>
<td>Section-06</td>
<td>0.000</td>
<td>-</td>
</tr>
<tr>
<td>Section-07</td>
<td>0.234</td>
<td>9.62</td>
</tr>
<tr>
<td>Section-08</td>
<td>0.571</td>
<td>3.94</td>
</tr>
</tbody>
</table>

(1) Ratio = Allowable Stress / Maximum Stress
# Table 3.8A-11

**Shear Reinforcement and Margins of Safety for RCB Basemat**

<table>
<thead>
<tr>
<th>Design Section</th>
<th>Required Reinforcement (in²/ft)</th>
<th>Provided Reinforcement (in²/ft)</th>
<th>Ratio&lt;sup&gt;(1)&lt;/sup&gt;</th>
<th>Remark&lt;sup&gt;(2)&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>Section-01</td>
<td>0.2932</td>
<td>0.3514</td>
<td>1.20</td>
<td></td>
</tr>
<tr>
<td>Section-02</td>
<td>0.5123</td>
<td>0.7029</td>
<td>1.37</td>
<td></td>
</tr>
<tr>
<td>Section-03</td>
<td>0.0896</td>
<td>0.1975</td>
<td>2.20</td>
<td></td>
</tr>
<tr>
<td>Section-04</td>
<td>0.0012</td>
<td>0.1975</td>
<td>164.58&lt;sup&gt;(2)&lt;/sup&gt;</td>
<td></td>
</tr>
<tr>
<td>Section-05</td>
<td>0.1029</td>
<td>0.1975</td>
<td>1.92</td>
<td></td>
</tr>
<tr>
<td>Section-06</td>
<td>0.3549</td>
<td>0.3950</td>
<td>1.11</td>
<td></td>
</tr>
<tr>
<td>Section-07</td>
<td>0.1444</td>
<td>0.1975</td>
<td>1.37</td>
<td></td>
</tr>
<tr>
<td>Section-08</td>
<td>0.3177</td>
<td>0.3950</td>
<td>1.24</td>
<td></td>
</tr>
</tbody>
</table>

<sup>(1)</sup> Ratio = Provided Reinforcement / Required Reinforcement

<sup>(2)</sup> This section is provided a minimum reinforcement of NI basemat.
# Summary of the Flexural Reinforcement of the RCB Basemat

<table>
<thead>
<tr>
<th>Design Section</th>
<th>EW or Radial Direction</th>
<th>NS or Hoop Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Top</td>
<td>Bottom</td>
</tr>
<tr>
<td>Section-01</td>
<td>1-#18@0.8°</td>
<td>2-#18@0.8°</td>
</tr>
<tr>
<td>Section-02</td>
<td>1-#18@0.4°</td>
<td>3-#14@0.8°</td>
</tr>
<tr>
<td>Section-03</td>
<td>3-#18@12&quot;</td>
<td>5-#18@6&quot;</td>
</tr>
<tr>
<td>Section-04</td>
<td>1-#18@0.4°</td>
<td>5-#18@6&quot;</td>
</tr>
<tr>
<td>Section-05</td>
<td>1-#18@0.4°</td>
<td>5-#18@6&quot;</td>
</tr>
<tr>
<td>Section-06</td>
<td>3-#18@12&quot;</td>
<td>1-#18@12&quot;</td>
</tr>
<tr>
<td>Section-07</td>
<td>3-#18@12&quot;</td>
<td>1-#18@12&quot;</td>
</tr>
<tr>
<td>Section-08</td>
<td>3-#18@12&quot;</td>
<td>1-#18@12&quot;</td>
</tr>
</tbody>
</table>
# Summary of the Shear Reinforcement of the RCB Basemat

<table>
<thead>
<tr>
<th>Design Section</th>
<th>Shear Reinforcement</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>EW or Radial Direction</td>
<td>NS or Hoop Direction</td>
<td></td>
</tr>
<tr>
<td>Section-01</td>
<td>1-#8@12”</td>
<td>1-#8@1.6°</td>
<td></td>
</tr>
<tr>
<td>Section-02</td>
<td>2-#8@12”</td>
<td>2-#8@1.6°</td>
<td></td>
</tr>
<tr>
<td>Section-03</td>
<td>1-#8@24”</td>
<td>1-#8@24”</td>
<td></td>
</tr>
<tr>
<td>Section-04</td>
<td>1-#8@24”</td>
<td>1-#8@24”</td>
<td></td>
</tr>
<tr>
<td>Section-05</td>
<td>1-#8@24”</td>
<td>1-#8@24”</td>
<td></td>
</tr>
<tr>
<td>Section-06</td>
<td>1-#8@24”</td>
<td>1-#8@24”</td>
<td></td>
</tr>
<tr>
<td>Section-07</td>
<td>1-#8@24”</td>
<td>1-#8@24”</td>
<td></td>
</tr>
<tr>
<td>Section-08</td>
<td>1-#8@12”</td>
<td>1-#8@24”</td>
<td></td>
</tr>
</tbody>
</table>
# Factor of Safety for Basemat Stability

<table>
<thead>
<tr>
<th>Load Combination</th>
<th>Overturning</th>
<th>Sliding</th>
<th>Flotation</th>
</tr>
</thead>
<tbody>
<tr>
<td>D + H_e + W</td>
<td>1.5</td>
<td>1.5</td>
<td>-</td>
</tr>
<tr>
<td>D + H_e + E_s</td>
<td>1.1</td>
<td>1.1</td>
<td>-</td>
</tr>
<tr>
<td>D + H_e + W_t</td>
<td>1.1</td>
<td>1.1</td>
<td>-</td>
</tr>
<tr>
<td>D + H_s</td>
<td>-</td>
<td>-</td>
<td>1.1</td>
</tr>
</tbody>
</table>

D = Dead load  
H_e = Static and dynamic lateral and vertical earth pressure including buoyant effect of normal design groundwater level  
H_s = Buoyant force of the design basis flood  
W = Wind load  
W_t = Tornado or hurricane load  
E_s = Safe shutdown earthquake
Table 3.8A-15

Results on Factor of Safety for Basemat Stability

<table>
<thead>
<tr>
<th>NI Common Basemat</th>
<th>FOS(1)</th>
<th>Allowable FOS</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overturning by Wind</td>
<td>16.46</td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td>Overturning by Earthquake</td>
<td>1.24</td>
<td>1.1</td>
<td></td>
</tr>
<tr>
<td>Sliding by Wind</td>
<td>8.30</td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td>Sliding by Earthquake</td>
<td>1.25</td>
<td>1.1</td>
<td>Time history method</td>
</tr>
<tr>
<td>Flotation</td>
<td>3.39</td>
<td>1.10</td>
<td></td>
</tr>
</tbody>
</table>

FOS = Factor of Safety
### Ground Contact Ratios of NI Common Basemat

<table>
<thead>
<tr>
<th>Site Profile</th>
<th>Concrete Stiffness</th>
<th>Critical Load Combination</th>
<th>Ground Contact Ratio (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>S1</td>
<td>Uncracked</td>
<td>1.0·D+1.0·SLL+1.0·Lh+1.0·SSE\textsubscript{EW}+1.0·SSE\textsubscript{NS}+1.0·SSE\textsubscript{VT}</td>
<td>95.74</td>
</tr>
<tr>
<td></td>
<td>Cracked</td>
<td>1.0·D+1.0·SLL+1.0·Lh+1.0·SSE\textsubscript{EW}+1.0·SSE\textsubscript{NS}+1.0·SSE\textsubscript{VT}</td>
<td>95.62</td>
</tr>
<tr>
<td>S4</td>
<td>Uncracked</td>
<td>1.0·D+1.0·SLL+1.0·Lh-1.0·SSE\textsubscript{EW}+1.0·SSE\textsubscript{NS}+1.0·SSE\textsuperscript{VT}</td>
<td>92.11</td>
</tr>
<tr>
<td></td>
<td>Cracked</td>
<td>1.0·D+1.0·SLL-1.0·Lh-1.0·SSE\textsubscript{EW}-1.0·SSE\textsubscript{NS}+1.0·SSE\textsubscript{VT}</td>
<td>92.11</td>
</tr>
<tr>
<td>S8</td>
<td>Uncracked</td>
<td>1.0·D+1.0·SLL+1.0·Lh+1.0·SSE\textsubscript{EW}+1.0·SSE\textsubscript{NS}+1.0·SSE\textsubscript{VT}</td>
<td>85.40</td>
</tr>
<tr>
<td></td>
<td>Cracked</td>
<td>1.0·D+1.0·SLL+1.0·Lh-1.0·SSE\textsubscript{EW}-1.0·SSE\textsubscript{NS}+1.0·SSE\textsubscript{VT}</td>
<td>88.90</td>
</tr>
</tbody>
</table>

- D = Dead load
- SLL = Seismic live load (25% of live loads)
- Lh = Buoyancy load due to groundwater
### Table 3.8A-17

NI Common Basemat Differential Settlement According to Site Profiles (Static)

<table>
<thead>
<tr>
<th>Section</th>
<th>Node #1</th>
<th>Node #2</th>
<th>Distance (ft)</th>
<th>Differential Settlement (inches)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Soil #1</td>
</tr>
<tr>
<td>AB1</td>
<td>26,810</td>
<td>27,829</td>
<td>48.58</td>
<td>0.197</td>
</tr>
<tr>
<td>AB2</td>
<td>27,829</td>
<td>29,466</td>
<td>46.26</td>
<td>0.206</td>
</tr>
<tr>
<td>AB3</td>
<td>29,466</td>
<td>28,901</td>
<td>46.59</td>
<td>0.189</td>
</tr>
<tr>
<td>AB4</td>
<td>28,901</td>
<td>1,367</td>
<td>44.70</td>
<td>0.209</td>
</tr>
<tr>
<td>AB5</td>
<td>26,811</td>
<td>27,246</td>
<td>48.73</td>
<td>0.067</td>
</tr>
<tr>
<td>AB6</td>
<td>27,246</td>
<td>26,610</td>
<td>44.08</td>
<td>0.074</td>
</tr>
<tr>
<td>AB7</td>
<td>26,610</td>
<td>27,669</td>
<td>41.54</td>
<td>0.087</td>
</tr>
<tr>
<td>AB8</td>
<td>27,669</td>
<td>790</td>
<td>39.68</td>
<td>0.123</td>
</tr>
<tr>
<td>AB9</td>
<td>26,620</td>
<td>28,027</td>
<td>48.73</td>
<td>0.001</td>
</tr>
<tr>
<td>AB10</td>
<td>28,027</td>
<td>26,667</td>
<td>44.08</td>
<td>0.019</td>
</tr>
<tr>
<td>AB11</td>
<td>26,667</td>
<td>27,610</td>
<td>41.54</td>
<td>0.010</td>
</tr>
<tr>
<td>AB12</td>
<td>27,610</td>
<td>822</td>
<td>39.68</td>
<td>0.059</td>
</tr>
<tr>
<td>AB13</td>
<td>26,826</td>
<td>27,117</td>
<td>48.58</td>
<td>0.166</td>
</tr>
<tr>
<td>AB14</td>
<td>27,117</td>
<td>29,708</td>
<td>46.26</td>
<td>0.180</td>
</tr>
<tr>
<td>AB15</td>
<td>29,708</td>
<td>30,238</td>
<td>46.59</td>
<td>0.155</td>
</tr>
<tr>
<td>AB16</td>
<td>30,238</td>
<td>1,466</td>
<td>44.70</td>
<td>0.170</td>
</tr>
<tr>
<td>RCB1</td>
<td>5,929</td>
<td>18,822</td>
<td>46.06</td>
<td>0.072</td>
</tr>
<tr>
<td>RCB2</td>
<td>15,931</td>
<td>15,467</td>
<td>47.09</td>
<td>0.006</td>
</tr>
<tr>
<td>RCB3</td>
<td>6,135</td>
<td>14,571</td>
<td>46.06</td>
<td>0.025</td>
</tr>
<tr>
<td>RCB4</td>
<td>16,131</td>
<td>15,368</td>
<td>47.09</td>
<td>0.051</td>
</tr>
<tr>
<td><strong>Total Max. Differential Settlement</strong></td>
<td></td>
<td></td>
<td></td>
<td>0.209</td>
</tr>
</tbody>
</table>
Table 3.8A-18

Design Forces and Moments for PSW

Unit: kips/ft, kips-ft/ft

<table>
<thead>
<tr>
<th>Location</th>
<th>$M_\phi$</th>
<th>$M_\theta$</th>
<th>$N_\phi$</th>
<th>$N_\theta$</th>
<th>$Q_\phi$</th>
<th>$Q_\theta$</th>
<th>$Q_T$</th>
<th>$M_{\phi\theta}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>North Wall</td>
<td>188.9</td>
<td>92.1</td>
<td>110.6</td>
<td>78.3</td>
<td>250.9</td>
<td>-152.2</td>
<td>282.7</td>
<td>-319.2</td>
</tr>
<tr>
<td></td>
<td>(-202.4)</td>
<td>(-100.3)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>South Wall</td>
<td>698.2</td>
<td>456.5</td>
<td>177.3</td>
<td>212.8</td>
<td>-268.7</td>
<td>-187.8</td>
<td>173.0</td>
<td>-359.4</td>
</tr>
<tr>
<td></td>
<td>(-619.9)</td>
<td>(-466.2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>East Wall</td>
<td>211.5</td>
<td>104.1</td>
<td>76.5</td>
<td>92.2</td>
<td>-202.9</td>
<td>-102.0</td>
<td>209.5</td>
<td>-190.9</td>
</tr>
<tr>
<td></td>
<td>(-192.9)</td>
<td>(-112.3)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Where,

- $M_\phi$: Meridional Moment around Horizontal Axis
- $M_\theta$: Hoop Moment around Vertical Axis
- $N_\phi$: Meridional Axial Force (+: Tension, -: Compression)
- $N_\theta$: Hoop Axial Force (+: Tension, -: Compression)
- $Q_\phi$: Meridional Transverse Shear Force
- $Q_\theta$: Hoop Transverse Shear Force
- $Q_T$: Tangential Shear Force (In-plane Shear Force)
- $M_{\phi\theta}$: Torsion Moment
Table 3.8A-19

Design Forces and Moments for IRWST

Unit: kips/ft, kips-ft/ft

<table>
<thead>
<tr>
<th>Location</th>
<th>( M_\phi )</th>
<th>( M_\theta )</th>
<th>( N_\phi )</th>
<th>( N_\theta )</th>
<th>( Q_\phi )</th>
<th>( Q_\theta )</th>
<th>( Q_T )</th>
<th>( M_{\phi \theta} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top Slab</td>
<td>60.2 (-19.7)</td>
<td>12.6 (-9.3)</td>
<td>45.5</td>
<td>15.4</td>
<td>17.5</td>
<td>15.7</td>
<td>16.3</td>
<td>-5.6</td>
</tr>
<tr>
<td>Outer Wall</td>
<td>50.3 (-51.7)</td>
<td>11.6 (-7.5)</td>
<td>-65.3</td>
<td>70.2</td>
<td>47.8</td>
<td>3.7</td>
<td>10.5</td>
<td>-4.0</td>
</tr>
</tbody>
</table>

Where,

- \( M_\phi \): Meridional Moment around Horizontal Axis
- \( M_\theta \): Hoop Moment around Vertical Axis
- \( N_\phi \): Meridional Axial Force (+: Tension, -: Compression)
- \( N_\theta \): Hoop Axial Force (+: Tension, -: Compression)
- \( Q_\phi \): Meridional Transverse Shear Force
- \( Q_\theta \): Hoop Transverse Shear Force
- \( Q_T \): Tangential Shear Force (In-plane Shear Force)
- \( M_{\phi \theta} \): Torsion Moment
Table 3.8A-20

Design Forces and Moments for SSW

Unit: kips/ft, kips-ft/ft

<table>
<thead>
<tr>
<th>Type</th>
<th>(M_\phi)</th>
<th>(M_\theta)</th>
<th>(N_\phi)</th>
<th>(N_\theta)</th>
<th>(Q_\phi)</th>
<th>(Q_\theta)</th>
<th>(Q_T)</th>
<th>(M_{\phi\theta})</th>
</tr>
</thead>
<tbody>
<tr>
<td>SSW (1)</td>
<td>299.6</td>
<td>283.6</td>
<td>159.5</td>
<td>235.1</td>
<td>-64.5</td>
<td>-65.2</td>
<td>-227.9</td>
<td>115.1</td>
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<tr>
<td></td>
<td>(-227.8)</td>
<td>(-176.4)</td>
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<tr>
<td>RFP (2)</td>
<td>1,479.3</td>
<td>1,009.0</td>
<td>256.9</td>
<td>382.9</td>
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<td>412.1</td>
<td>-341.9</td>
<td>662.1</td>
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<td></td>
<td>(-1,242.2)</td>
<td>(-1,026.6)</td>
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<td></td>
</tr>
<tr>
<td>RFP (3)</td>
<td>138.7</td>
<td>330.6</td>
<td>48.1</td>
<td>505.8</td>
<td>-50.5</td>
<td>103.6</td>
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<td>(-325.9)</td>
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<tr>
<td>SG (4)</td>
<td>473.8</td>
<td>490.9</td>
<td>353.1</td>
<td>249.0</td>
<td>129.6</td>
<td>-98.1</td>
<td>-184.2</td>
<td>276.1</td>
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<tr>
<td></td>
<td>(-337.7)</td>
<td>(-428.5)</td>
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</tr>
<tr>
<td>SG (5)</td>
<td>1,026.8</td>
<td>758.7</td>
<td>198.3</td>
<td>250.4</td>
<td>181.2</td>
<td>-147.1</td>
<td>238.3</td>
<td>-425.6</td>
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<tr>
<td></td>
<td>(-880.7)</td>
<td>(-708.3)</td>
<td></td>
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<td></td>
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<tr>
<td>PZR (6)</td>
<td>79.3</td>
<td>318.7</td>
<td>216.3</td>
<td>230.9</td>
<td>-59.0</td>
<td>-109.1</td>
<td>165.4</td>
<td>78.8</td>
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<tr>
<td></td>
<td>(-75.9)</td>
<td>(-308.8)</td>
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<td></td>
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<td></td>
</tr>
</tbody>
</table>

(1) Secondary Shield Wall (Thickness 4 ft)
(2) South/North Wall of Refueling pool (thickness 6 ft 2 in)
(3) West Wall of Refueling pool (Thickness 5 ft)
(4) Circular Wall of Steam Generator (SG) Enclosure (Thickness 4 ft)
(5) Straight Wall of Steam Generator (SG) Enclosure (Thickness 5 ft)
(6) Pressurizer (PZR) Enclosure Wall (Thickness 2 ft 9 in)

Where:

- \(M_\phi\): Vertical Moment around Horizontal Axis
- \(M_\theta\): Horizontal Moment around Vertical Axis
- \(N_\phi\): Vertical Axial Force (+: Tension, -: Compression)
- \(N_\theta\): Horizontal Axial Force (+: Tension, -: Compression)
- \(Q_\phi\): Vertical Transverse Shear Force
- \(Q_\theta\): Horizontal Transverse Shear Force
- \(Q_T\): In-plane Shear Force
- \(M_{\phi\theta}\): Torsion Moment
## Table 3.8A-21

### Typical Rebar Arrangement for PSW

<table>
<thead>
<tr>
<th>Location</th>
<th>Meridional Direction</th>
<th>Hoop Direction</th>
<th>Shear Reinforcement</th>
</tr>
</thead>
<tbody>
<tr>
<td>North Wall</td>
<td>Inside : #18 &amp; #18 @ 4.5°</td>
<td>Inside : #18 &amp; #18 @ 12”</td>
<td>#8 @ 12” Vert. @ 4.5° Horiz.</td>
</tr>
<tr>
<td></td>
<td>Outside : #18 &amp; #18 @ 12”</td>
<td>Outside : #18 &amp; #18 @ 12”</td>
<td></td>
</tr>
<tr>
<td>East Wall</td>
<td>Inside : 2-#18(Bundled) &amp; #14 @ 4.5°</td>
<td>Inside : #18 &amp; #18 @ 12”</td>
<td>#8 @ 12” Vert. @ 4.5° Horiz.</td>
</tr>
<tr>
<td></td>
<td>Outside : #18 &amp; #18 &amp; #14 @ 12”</td>
<td>Outside : #18 &amp; #18 @ 12”</td>
<td></td>
</tr>
<tr>
<td>South Wall</td>
<td>Inside : #18 &amp; #14 @ 4.5°</td>
<td>Inside : #18 &amp; #14 @ 12”</td>
<td>#8 @ 12” Vert. @ 4.5° Horiz.</td>
</tr>
<tr>
<td></td>
<td>Outside : #18 &amp; #14 @ 12”</td>
<td>Outside : #18 &amp; #14 @ 12”</td>
<td></td>
</tr>
</tbody>
</table>
Table 3.8A-22

Margins of Safety for Primary Shield Wall

<table>
<thead>
<tr>
<th>Location</th>
<th>Meridional/Vertical Direction</th>
<th>Hoop/Horizontal Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Reinforcement</td>
<td>Reinforcement</td>
</tr>
<tr>
<td></td>
<td>Required (in²)</td>
<td>Provided</td>
</tr>
<tr>
<td>North Wall</td>
<td>6.96</td>
<td>#18 &amp; #18@4.5° (8.00in²)</td>
</tr>
<tr>
<td>East Wall</td>
<td>7.75</td>
<td>2-#18 &amp; #14@4.5° (10.38in²)</td>
</tr>
<tr>
<td>South Wall</td>
<td>5.18</td>
<td>#18 &amp; #14@4.5° (6.28in²)</td>
</tr>
</tbody>
</table>

(¹) Ratio = Provided Rebar / Required Rebar
Table 3.8A-23

Typical Rebar Arrangement for IRWST

<table>
<thead>
<tr>
<th>Location</th>
<th>Meridional/Vertical Direction</th>
<th>Hoop/Horizontal Direction</th>
<th>Shear Reinforcement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top Slab</td>
<td>#11 @ 0.9° Each Face</td>
<td>#11 @ 12&quot; Each Face</td>
<td>-</td>
</tr>
<tr>
<td>Outer Wall</td>
<td>#11 @ 0.9° Each Face</td>
<td>#11 @ 12&quot; Each Face</td>
<td>#6 @ 24&quot; Vert. @ 0.9° Horiz.</td>
</tr>
</tbody>
</table>
## Table 3.8A-24

### Typical Rebar Arrangement for SSW

<table>
<thead>
<tr>
<th>Location</th>
<th>Meridional/Vertical Direction</th>
<th>Hoop/Horizontal Direction</th>
<th>Shear Reinforcement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Secondary Shield Wall</td>
<td>#18 &amp; #14 @ 0.9° Each Face</td>
<td>#18 &amp; #18 @ 12” Each Face</td>
<td>#7 @ 12” Vert. @ 0.9° Horiz.</td>
</tr>
<tr>
<td>Refueling Pool Wall</td>
<td>2-#18(Bundled) &amp; 2-#18(Bundled) @ 12” Each Face</td>
<td>2-#18(Bundled) &amp; 2-#18(Bundled) @ 12” Each Face</td>
<td>2-#8 @ 12” Vert. @ 6” Horiz.</td>
</tr>
<tr>
<td>South/North</td>
<td>#14 &amp; #11 @ 12” Each Face</td>
<td>#18 #14 &amp; #18 @ 12” Each Face</td>
<td>#8 @ 12” Vert. @ 12” Horiz.</td>
</tr>
<tr>
<td>West</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SG Enclosure Wall</td>
<td>#18 #14 &amp; #18 @ 0.9° Each Face</td>
<td>#18 #14 &amp; #18 @ 12” Each Face</td>
<td>2-#7 @ 12” Vert. @ 0.9° Horiz.</td>
</tr>
<tr>
<td>Circular</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Straight</td>
<td>2-#18(Bundled) &amp; 2-#18(Bundled) @ 12” Each Face</td>
<td>2-#18 &amp; #18 @ 12” Each Face</td>
<td>2-#8 @ 12” Vert. @ 12” Hor.</td>
</tr>
<tr>
<td>PZR Enclosure Wall</td>
<td>2-#14(Bundled) &amp; #14 @ 12” Each Face</td>
<td>#18 #14 &amp; #18@ 12” Each Face</td>
<td>2-#8 @ 12” Vert. @ 12” Hor.</td>
</tr>
</tbody>
</table>
### Table 3.8A-25

Margins of Safety for Secondary Shield Wall

<table>
<thead>
<tr>
<th>Location</th>
<th>Meridional/Vertical Direction</th>
<th>Hoop/Horizontal Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Reinforcement</td>
<td>Required (in²)</td>
</tr>
<tr>
<td>SSW</td>
<td>#18 &amp; #14@0.9°</td>
<td>6.68</td>
</tr>
<tr>
<td>RPW North/South</td>
<td>2-#18 &amp; 2-#18@12&quot;</td>
<td>14.06</td>
</tr>
<tr>
<td>RPW West</td>
<td>#14 &amp; #11@12&quot;</td>
<td>3.00</td>
</tr>
<tr>
<td>SG Circular</td>
<td>#18@0.9° &amp; #18@12&quot;</td>
<td>9.99</td>
</tr>
<tr>
<td>SG Straight</td>
<td>2-#18 &amp; 2-#18@12&quot;</td>
<td>12.41</td>
</tr>
<tr>
<td>PZR</td>
<td>2-#14 &amp; #14@12&quot;</td>
<td>5.50</td>
</tr>
</tbody>
</table>

(1) Ratio = Provided Rebar / Required Rebar
## Table 3.8A-26

Enveloped Design Forces of the AB Basemat

<table>
<thead>
<tr>
<th>Element Set No.</th>
<th>Governing Load Combinations</th>
<th>(N_x) kip/ft</th>
<th>(N_y) kip/ft</th>
<th>(M_x) kip-ft/ft</th>
<th>(M_y) kip-ft/ft</th>
<th>(V_{xz}) kip/ft</th>
<th>(V_{yz}) kip/ft</th>
<th>(V_{in_x}) kip/ft</th>
<th>(V_{in_y}) kip/ft</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>LC11</td>
<td>362.0</td>
<td>500.3</td>
<td>901.1</td>
<td>1,514.5</td>
<td>95.3</td>
<td>159.3</td>
<td>-712.0</td>
<td>-712.0</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>LC9</td>
<td>298.3</td>
<td>409.5</td>
<td>930.4</td>
<td>1,319.8</td>
<td>81.6</td>
<td>210.1</td>
<td>634.7</td>
<td>634.7</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>LC15</td>
<td>-471.4</td>
<td>142.7</td>
<td>1,969.0</td>
<td>1,211.5</td>
<td>216.7</td>
<td>37.0</td>
<td>304.1</td>
<td>304.1</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>LC13</td>
<td>-479.6</td>
<td>112.4</td>
<td>2,268.5</td>
<td>1,428.9</td>
<td>234.8</td>
<td>50.9</td>
<td>-332.6</td>
<td>-332.6</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>LC6</td>
<td>-482.6</td>
<td>262.4</td>
<td>2,341.8</td>
<td>1,406.9</td>
<td>285.4</td>
<td>128.6</td>
<td>325.2</td>
<td>321.9</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>LC13</td>
<td>-465.7</td>
<td>229.8</td>
<td>2,637.7</td>
<td>1,557.6</td>
<td>304.3</td>
<td>146.6</td>
<td>-353.0</td>
<td>-356.8</td>
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</tr>
<tr>
<td>7</td>
<td>LC6</td>
<td>258.9</td>
<td>434.8</td>
<td>1,066.2</td>
<td>330.4</td>
<td>138.7</td>
<td>49.7</td>
<td>-46.9</td>
<td>-86.7</td>
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</tr>
<tr>
<td>8</td>
<td>LC11</td>
<td>1,151.2</td>
<td>521.0</td>
<td>1,148.5</td>
<td>1,503.5</td>
<td>101.8</td>
<td>155.0</td>
<td>-278.7</td>
<td>-208.3</td>
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<tr>
<td>9</td>
<td>LC9</td>
<td>1,006.8</td>
<td>435.2</td>
<td>1,220.5</td>
<td>1,309.9</td>
<td>72.3</td>
<td>156.7</td>
<td>244.4</td>
<td>182.9</td>
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<tr>
<td>10</td>
<td>LC14</td>
<td>-195.4</td>
<td>-54.9</td>
<td>612.2</td>
<td>584.5</td>
<td>134.4</td>
<td>99.2</td>
<td>-454.6</td>
<td>-454.6</td>
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</tr>
<tr>
<td>11</td>
<td>LC9</td>
<td>-450.1</td>
<td>-126.2</td>
<td>725.8</td>
<td>1,181.7</td>
<td>38.8</td>
<td>205.9</td>
<td>446.6</td>
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<tr>
<td>12</td>
<td>LC9</td>
<td>-274.3</td>
<td>193.8</td>
<td>2,082.1</td>
<td>1,667.6</td>
<td>595.1</td>
<td>122.4</td>
<td>-574.2</td>
<td>-586.4</td>
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</tr>
<tr>
<td>13</td>
<td>LC6</td>
<td>110.8</td>
<td>-5.4</td>
<td>370.0</td>
<td>1,196.3</td>
<td>85.5</td>
<td>153.3</td>
<td>-52.7</td>
<td>-21.0</td>
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</tr>
<tr>
<td>14</td>
<td>LC15</td>
<td>-200.0</td>
<td>57.5</td>
<td>889.4</td>
<td>358.0</td>
<td>91.2</td>
<td>134.4</td>
<td>-352.5</td>
<td>-352.5</td>
<td></td>
</tr>
<tr>
<td>15</td>
<td>LC11</td>
<td>-466.1</td>
<td>-765.9</td>
<td>575.6</td>
<td>1,231.8</td>
<td>126.9</td>
<td>187.5</td>
<td>-225.0</td>
<td>-95.6</td>
<td></td>
</tr>
</tbody>
</table>

(1) The location of the element sets is shown in Table 3.8A-27. X-axis is in EW direction, and Y-axis is in NS direction. Governing load combinations are as follows;

- **LC6**: 1.4D+1.7L+1.4Lₙₜ+1.0F
- **LC9**: 1.0D+1.0L+1.0Lₙₜ+1.0F+1.0Pₙ+1.0Yₙₜ+1.0E₀₂
- **LC11**: 1.0D+1.0L+1.0Lₙₜ+1.0F+1.0Pₙ+1.0Yₙₜ+1.0E₀₄
- **LC13**: 1.0D+1.0L+1.0Lₙₜ+1.0F+1.0Pₙ+1.0Yₙₜ+1.0E₀₆
- **LC14**: 1.0D+1.0L+1.0Lₙₜ+1.0F+1.0Pₙ+1.0Yₙₜ+1.0E₀₇
- **LC15**: 1.0D+1.0L+1.0Lₙₜ+1.0F+1.0Pₙ+1.0Yₙₜ+1.0E₀₈
### Table 3.8A-27

Required Reinforcement and Margins of Safety for the AB Basemat

<table>
<thead>
<tr>
<th>Element Set No.</th>
<th>Required Top and Bottom Bar</th>
<th>Shear Bar</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>EW-Direction</td>
<td>NS-Direction</td>
</tr>
<tr>
<td></td>
<td>Required in²</td>
<td>Provided</td>
</tr>
<tr>
<td>1</td>
<td>9.19</td>
<td>3-#18@12&quot;</td>
</tr>
<tr>
<td>2</td>
<td>7.4</td>
<td>2-#18@12&quot;</td>
</tr>
<tr>
<td>3</td>
<td>5.28</td>
<td>3-#14@12&quot;</td>
</tr>
<tr>
<td>4</td>
<td>6.11</td>
<td>3-#14@12&quot;</td>
</tr>
<tr>
<td>5</td>
<td>5.25</td>
<td>3-#14@12&quot;</td>
</tr>
<tr>
<td>6</td>
<td>7.15</td>
<td>2-#18@12&quot;</td>
</tr>
<tr>
<td>7</td>
<td>4.79</td>
<td>3-#14@12&quot;</td>
</tr>
<tr>
<td>8</td>
<td>9.57</td>
<td>3-#18@12&quot;</td>
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<tr>
<td>9</td>
<td>11.03</td>
<td>3-#18@12&quot;</td>
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<td>10</td>
<td>3.71</td>
<td>3-#11@12&quot;</td>
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<tr>
<td>11</td>
<td>3.78</td>
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<td>12</td>
<td>8.42</td>
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<td>2.59</td>
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<tr>
<td>14</td>
<td>3.2</td>
<td>3-#11@12&quot;</td>
</tr>
<tr>
<td>15</td>
<td>2.59</td>
<td>2-#11@12&quot;</td>
</tr>
</tbody>
</table>

(1) Ratio = Provided Reinforcement / Required Reinforcement

(2) The element sets of the auxiliary building basemat are composed as following:
Table 3.8A-28 (1 of 2)

<table>
<thead>
<tr>
<th>Element Set No.</th>
<th>Zone Description</th>
<th>Governing Load Combinations</th>
<th>N11 kip/ft</th>
<th>N22 kip/ft</th>
<th>M11 kip-ft/ft</th>
<th>M22 kip-ft/ft</th>
<th>Qout kip/ft</th>
<th>N12 kip/ft</th>
</tr>
</thead>
<tbody>
<tr>
<td>North wall of north MSIV house</td>
<td>1</td>
<td>55'-0&quot; to 100'-0&quot;</td>
<td>LC15C</td>
<td>91.8</td>
<td>159.9</td>
<td>467.4</td>
<td>271.0</td>
<td>235.0</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>100'-0&quot; to 120'-0&quot;</td>
<td>LC15C</td>
<td>30.6</td>
<td>87.6</td>
<td>152.0</td>
<td>147.1</td>
<td>70.9</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>120'-0&quot; to 137'-6&quot;</td>
<td>LC15A</td>
<td>68.9</td>
<td>29.5</td>
<td>128.4</td>
<td>72.8</td>
<td>34.4</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>137'-6&quot; to 174'-0&quot;</td>
<td>LC15A</td>
<td>112.8</td>
<td>7.5</td>
<td>55.1</td>
<td>65.5</td>
<td>2.1</td>
</tr>
<tr>
<td>North wall of north AFWST</td>
<td>1</td>
<td>100'-0&quot; to 137'-6&quot;</td>
<td>LC15B</td>
<td>-53.9</td>
<td>-144.0</td>
<td>74.3</td>
<td>124.5</td>
<td>9</td>
</tr>
<tr>
<td>West wall of MCR</td>
<td>1</td>
<td>55'-0&quot; to 100'-0&quot;</td>
<td>LC15B</td>
<td>14.3</td>
<td>109.0</td>
<td>73.3</td>
<td>70.0</td>
<td>54.6</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>100'-0&quot; to 137'-6&quot;</td>
<td>LC15C</td>
<td>-59.6</td>
<td>-277.4</td>
<td>76.8</td>
<td>67.7</td>
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<tr>
<td></td>
<td>3</td>
<td>137'-6&quot; to 156'-0&quot;</td>
<td>LC15C</td>
<td>-196.9</td>
<td>-146.7</td>
<td>29.3</td>
<td>27.1</td>
<td>12.8</td>
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<tr>
<td></td>
<td>4</td>
<td>156'-0&quot; to 174'-0&quot;</td>
<td>LC15B</td>
<td>20.1</td>
<td>-15.4</td>
<td>28.9</td>
<td>51.8</td>
<td>39.6</td>
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<td>174'-0&quot; to 195'-0&quot;</td>
<td>LC15C</td>
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<td>44.7</td>
<td>73.9</td>
<td>17.5</td>
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<tr>
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<td>1</td>
<td>114'-0&quot; to 156'-0&quot;</td>
<td>LC15B</td>
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<td>288.4</td>
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<tr>
<td>East wall of FHA</td>
<td>1</td>
<td>156'-0&quot; to 174'-0&quot;</td>
<td>LC15C</td>
<td>72.4</td>
<td>-1.6</td>
<td>69.5</td>
<td>69.2</td>
<td>59.7</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>174'-0&quot; to 213'-6&quot;</td>
<td>LC15B</td>
<td>-68.2</td>
<td>-155.8</td>
<td>125.8</td>
<td>222.1</td>
<td>20.1</td>
</tr>
</tbody>
</table>

(1) 11 is in horizontal direction, and 22 is in vertical direction.
(2) Design forces and moments in this table are the enveloped values of the following load combinations:

- L.C.4 : 1.4 D + 1.7 (L + L_g_s) + 1.4 L_h + 1.4 R_o
- L.C.6 : 1.4 D + 1.7 (L + L_g_s) + 1.4 L_h + 1.7 W
- L.C.15A : 1.0 D + 1.0 L + 1.0 L_h + 1.0 L_g_s + 1.0 E_s + 1.0 L_g_d
- L.C.15B : 1.0 D + 1.0 L + 1.0 L_h + 1.0 L_g_s - 1.0 E_s + 1.0 L_g_d
- L.C.15C : 0.9 D + 1.0 L_h + 1.0 L_g_s + 1.0 E_s + 1.0 L_g_d
- L.C.15D : 0.9 D + 1.0 L_h + 1.0 L_g_s - 1.0 E_s + 1.0 L_g_d

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Rev. 3
(3) The sign convention for design forces and moments is as follows.

- **N11, N22**: In-plane axial forces, (+) tension, (-) compression
- **M11, M22**: Out-of plane moments
  
  \[
  M_{11} = \max[(+M_{11} + |M_{12}|) \text{ or } \min[(-)M_{11} - |M_{12}|)]
  \]
  
  \[
  M_{22} = \max[(+M_{22} + |M_{12}|) \text{ or } \min[(-)M_{22} - |M_{12}|)]
  \]
- **Qout**: Transverse shear forces
  
  **Q13**: Transverse shear in the axis 1-3 plane
  
  **Q23**: Transverse shear in the axis 2-3 plane

  \[
  Q_{out} = \max[Q_{13}, Q_{23}] \text{ or } \min[Q_{13}, Q_{23}]
  \]
- **N12**: In-plane shear forces
## Required Reinforcement and Margins of Safety for the AB Shear Wall

<table>
<thead>
<tr>
<th>Critical Section</th>
<th>Zone</th>
<th>Elevation</th>
<th>Horizontal</th>
<th>Vertical</th>
<th>Shear</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Required in^2</td>
<td>Provided</td>
<td>Ratio(1)</td>
<td>Required in^2</td>
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<tr>
<td>North wall of north MSIV house</td>
<td>1</td>
<td>55'-0&quot; to 100'-0&quot;</td>
<td>5.85</td>
<td>2-#14@9&quot;</td>
<td>1.03</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>100'-0&quot; to 120'-0&quot;</td>
<td>3.49</td>
<td>3-#14@9&quot;</td>
<td>1.19</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>120'-0&quot; to 137'-6&quot;</td>
<td>3.23</td>
<td>3-#14@9&quot;</td>
<td>1.29</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>137'-6&quot; to 174'-0&quot;</td>
<td>3.59</td>
<td>3-#14@9&quot;</td>
<td>2.51</td>
</tr>
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<td>North wall of north AFWST</td>
<td>1</td>
<td>100'-0&quot; to 137'-6&quot;</td>
<td>3.16</td>
<td>2-#11@9&quot;</td>
<td>1.32</td>
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<tr>
<td>West wall of MCR</td>
<td>1</td>
<td>55'-0&quot; to 100'-0&quot;</td>
<td>3.99</td>
<td>2-#14@12&quot;</td>
<td>1.17</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>100'-0&quot; to 137'-6&quot;</td>
<td>2.59</td>
<td>2-#11@12&quot;</td>
<td>1.20</td>
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<tr>
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<td>3</td>
<td>137'-6&quot; to 156'-0&quot;</td>
<td>2.39</td>
<td>3-#11@9&quot;</td>
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<tr>
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<td>4</td>
<td>156'-0&quot; to 174'-0&quot;</td>
<td>1.28</td>
<td>#11@12&quot;</td>
<td>1.22</td>
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<tr>
<td></td>
<td>5</td>
<td>174'-0&quot; to 195'-0&quot;</td>
<td>1.42</td>
<td>#11@12&quot;</td>
<td>1.10</td>
</tr>
<tr>
<td>West wall of SFP</td>
<td>1</td>
<td>114'-0&quot; to 156'-0&quot;</td>
<td>4.79</td>
<td>2-#14@9&quot;</td>
<td>1.25</td>
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<tr>
<td>East wall of FHA area</td>
<td>1</td>
<td>156'-0&quot; to 174'-0&quot;</td>
<td>2.73</td>
<td>2-#11@9&quot;</td>
<td>1.52</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>174'-0&quot; to 213'-6&quot;</td>
<td>2.67</td>
<td>2-#10@9&quot;</td>
<td>1.27</td>
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</table>

(1) Ratio = Provided Reinforcement / Required Reinforcement
### Table 3.8A-30

**Enveloped Design Forces of the EDG Room Slab in AB**

<table>
<thead>
<tr>
<th>Location</th>
<th>Design Force and Moments</th>
<th>Critical Joint No.</th>
<th>Governing Load Combination</th>
<th>$M_{xx}$ (kip-ft/ft)</th>
<th>$M_{yy}$ (kip-ft/ft)</th>
<th>$M_{xy}$ (kip-ft/ft)</th>
<th>$V_{xx}$ (kip/ft)</th>
<th>$V_{yy}$ (kip/ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EDG-1 zone at Elevation 100'-0&quot; (24&quot; thick)</td>
<td>Max. positive $M_{xx}$</td>
<td>195</td>
<td>LC15D</td>
<td>15.9</td>
<td>10.31</td>
<td>0.15</td>
<td>0.93</td>
<td>0.26</td>
</tr>
<tr>
<td></td>
<td></td>
<td>196</td>
<td>LC15D</td>
<td>14.55</td>
<td>9.53</td>
<td>1.66</td>
<td>0.83</td>
<td>-0.27</td>
</tr>
<tr>
<td></td>
<td>Max. negative $M_{xx}$</td>
<td>443</td>
<td>LC15A</td>
<td>-21.55</td>
<td>0.34</td>
<td>-0.72</td>
<td>-7.55</td>
<td>-0.11</td>
</tr>
<tr>
<td></td>
<td></td>
<td>450</td>
<td>LC15A</td>
<td>-19.11</td>
<td>1.34</td>
<td>-0.42</td>
<td>-6.21</td>
<td>-0.16</td>
</tr>
<tr>
<td></td>
<td>Max. positive $M_{yy}$</td>
<td>182</td>
<td>LC15D</td>
<td>12.12</td>
<td>11.58</td>
<td>0.50</td>
<td>0.68</td>
<td>0.08</td>
</tr>
<tr>
<td></td>
<td></td>
<td>188</td>
<td>LC15D</td>
<td>12.82</td>
<td>11.23</td>
<td>0.84</td>
<td>1.07</td>
<td>-0.10</td>
</tr>
<tr>
<td></td>
<td>Max. negative $M_{yy}$</td>
<td>264</td>
<td>LC15A</td>
<td>-5.99</td>
<td>-27.97</td>
<td>-0.79</td>
<td>0.54</td>
<td>5.18</td>
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<td>185</td>
<td>LC15A</td>
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<td>-27.13</td>
<td>-1.15</td>
<td>-1.83</td>
<td>-0.58</td>
</tr>
<tr>
<td></td>
<td>Max. $V_{xx}$</td>
<td>443</td>
<td>LC15A</td>
<td>-21.55</td>
<td>0.34</td>
<td>-0.72</td>
<td>-7.55</td>
<td>-0.11</td>
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<td>0.93</td>
<td>-6.75</td>
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<td>Max. $V_{yy}$</td>
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<td>1.00</td>
<td>7.56</td>
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<tr>
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<td>179</td>
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<td>-14.41</td>
<td>1.26</td>
<td>-0.73</td>
<td>7.18</td>
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<tr>
<td>EDG-2 zone at Elevation 100'-0&quot; (48&quot; thick)</td>
<td>Max. positive $M_{xx}$</td>
<td>1114</td>
<td>LC15D</td>
<td>27.51</td>
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<td>31.71</td>
<td>38.79</td>
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<td>791</td>
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<td>Max. negative $M_{xx}$</td>
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<td>LC15A</td>
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<td>-86.84</td>
<td>-23.22</td>
<td>8.08</td>
<td>13.89</td>
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<td></td>
<td>453</td>
<td>LC15A</td>
<td>-23.46</td>
<td>-32.54</td>
<td>1.37</td>
<td>-0.25</td>
<td>-40.74</td>
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<tr>
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<td>Max. positive $M_{yy}$</td>
<td>450</td>
<td>LC15D</td>
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<td>50.99</td>
<td>-2.00</td>
<td>3.34</td>
<td>2.98</td>
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<td>620</td>
<td>LC15D</td>
<td>16.12</td>
<td>46.93</td>
<td>-1.56</td>
<td>2.22</td>
<td>1.61</td>
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<tr>
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<td>Max. negative $M_{yy}$</td>
<td>434</td>
<td>LC15A</td>
<td>-29.26</td>
<td>-86.84</td>
<td>-23.22</td>
<td>8.08</td>
<td>13.89</td>
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<td>786</td>
<td>LC15A</td>
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<td>1.29</td>
<td>-0.22</td>
<td>-3.87</td>
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<td>Max. $V_{xx}$</td>
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<td>LC15D</td>
<td>27.51</td>
<td>14.68</td>
<td>31.71</td>
<td>38.79</td>
<td>3.68</td>
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<td>26.07</td>
<td>4.50</td>
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<tr>
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<td>Max. $V_{yy}$</td>
<td>453</td>
<td>LC15A</td>
<td>-23.46</td>
<td>-32.54</td>
<td>1.37</td>
<td>-0.25</td>
<td>-40.74</td>
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<td>-5.60</td>
<td>-0.84</td>
<td>-22.68</td>
</tr>
</tbody>
</table>

(1) X is in EW direction, and Y is in NS direction.

(2) Governing load combinations are as follows;
- LC15A : D+L+E_s (downward) + P_a (downward)
- LC15D : 0.9D+L+E_s (upward) + P_a (upward)

where, $P_a$: Compartment pressure due to HELB
Table 3.8A-31

Enveloped Design Forces of the SFP Slab in AB

<table>
<thead>
<tr>
<th>Location</th>
<th>Load Case</th>
<th>Design Force and Moments</th>
<th>Critical joint at model</th>
<th>N_x (kip/ft)</th>
<th>N_y (kip/ft)</th>
<th>M_xx (kip-ft/ft)</th>
<th>M_yy (kip-ft/ft)</th>
<th>V_out (kip/ft)</th>
<th>N_y (kip/ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SFP slab at Elevation 114'-0&quot;</td>
<td>Mechanical Load&lt;sup&gt;(1)&lt;/sup&gt;</td>
<td>Max. positive M_xx</td>
<td>42,941</td>
<td>34.5</td>
<td>22.8</td>
<td>142.4</td>
<td>176.3</td>
<td>210.2</td>
<td>-224.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Max. negative M_xx</td>
<td>42,944</td>
<td>377.8</td>
<td>375.3</td>
<td>127.9</td>
<td>77.9</td>
<td>130.1</td>
<td>-89.2</td>
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<tr>
<td></td>
<td></td>
<td>Max. positive M_yy</td>
<td>70,093</td>
<td>30.0</td>
<td>27.3</td>
<td>122.6</td>
<td>216.2</td>
<td>120.2</td>
<td>-133.6</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Max. negative M_yy</td>
<td>42,939</td>
<td>98.2</td>
<td>198.9</td>
<td>72.4</td>
<td>162.8</td>
<td>116.0</td>
<td>-90.4</td>
</tr>
<tr>
<td></td>
<td>Thermal Load&lt;sup&gt;(2)&lt;/sup&gt;</td>
<td>Max. positive M_xx</td>
<td>42,941</td>
<td>-408.1</td>
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<td>-1,098.5</td>
<td>-855.4</td>
<td>108.3</td>
<td>134.7</td>
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<td></td>
<td>Max. negative M_xx</td>
<td>42,944</td>
<td>-311.2</td>
<td>-501.5</td>
<td>-161.5</td>
<td>-379.0</td>
<td>51.5</td>
<td>337.7</td>
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<td>Max. positive M_yy</td>
<td>70,093</td>
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<td>-590.9</td>
<td>-7,724.5</td>
<td>-790.6</td>
<td>22.1</td>
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<td>Max. negative M_yy</td>
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<td>-1,222.5</td>
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<td>188.8</td>
<td>103.4</td>
</tr>
</tbody>
</table>

<sup>(1)</sup> Mechanical loads are member forces combining the results obtained from structural analysis with those obtained from hydrodynamic analysis at spent fuel pool.

<sup>(2)</sup> Thermal loads are member forces induced by temperature difference during abnormal operating conditions.
### Table 3.8A-32

Enveloped Design Forces of the MSE Slab in AB

<table>
<thead>
<tr>
<th>Location</th>
<th>Design Force and Moments</th>
<th>Critical joint at model</th>
<th>Load Combination</th>
<th>$M_{xx}$ (kip-ft/ft)</th>
<th>$M_{yy}$ (kip-ft/ft)</th>
<th>$M_{xy}$ (kip-ft/ft)</th>
<th>$V_{xx}$ (kip/ft)</th>
<th>$V_{yy}$ (kip/ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td>MSE slab at Elevation 157'6&quot;</td>
<td>Max. positive $M_{xx}$</td>
<td>619</td>
<td>LC15A</td>
<td>93.8</td>
<td>135.8</td>
<td>0.6</td>
<td>0.6</td>
<td>-6.9</td>
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<td></td>
<td>688</td>
<td>LC15A</td>
<td>88.7</td>
<td>125.0</td>
<td>-4.6</td>
<td>-7.3</td>
<td>-6.5</td>
</tr>
<tr>
<td></td>
<td>Max. negative $M_{xx}$</td>
<td>887</td>
<td>LC15A</td>
<td>-146.8</td>
<td>-29.8</td>
<td>-7.1</td>
<td>-8.1</td>
<td>3.2</td>
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<td>343</td>
<td>LC15A</td>
<td>-105.9</td>
<td>-45.0</td>
<td>4.5</td>
<td>-9.5</td>
<td>2.0</td>
</tr>
<tr>
<td></td>
<td>Max. positive $M_{yy}$</td>
<td>618</td>
<td>LC15A</td>
<td>88.6</td>
<td>138.6</td>
<td>-0.6</td>
<td>0.5</td>
<td>8.0</td>
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<tr>
<td></td>
<td></td>
<td>619</td>
<td>LC15A</td>
<td>93.8</td>
<td>135.8</td>
<td>0.6</td>
<td>0.6</td>
<td>-6.9</td>
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<td>Max. negative $M_{yy}$</td>
<td>1392</td>
<td>LC15A</td>
<td>-45.8</td>
<td>-123.9</td>
<td>2.3</td>
<td>1.6</td>
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<td>1265</td>
<td>LC15A</td>
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<td>-111.9</td>
<td>-1.4</td>
<td>-12.9</td>
<td>-67.9</td>
</tr>
<tr>
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<td>Max. $V_{xx}$</td>
<td>1557</td>
<td>LC15A</td>
<td>6.9</td>
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<td>-43.1</td>
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<td></td>
<td>412</td>
<td>LC15A</td>
<td>-9.4</td>
<td>28.5</td>
<td>13.8</td>
<td>39.2</td>
<td>-3.1</td>
</tr>
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<td>Max. $V_{yy}$</td>
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<td>LC15A</td>
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<td>-0.3</td>
<td>-60.1</td>
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<td></td>
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<td>LC15A</td>
<td>-35.8</td>
<td>-201.7</td>
<td>-6.2</td>
<td>10.7</td>
<td>-57.7</td>
</tr>
</tbody>
</table>

(1) X is in EW direction, and Y is in NS direction.

(2) Governing load combinations are as follows:
- LC15A : D+L+E_s (downward)+ P_a (downward)
- LC15D : 0.9D+L+E_s (upward)+ P_a (upward)
  where, P_a: Compartment pressure due to HELB
### Table 3.8A-33

Slab Reinforcement and Margins of Safety at Each Critical Section in AB

<table>
<thead>
<tr>
<th>Location</th>
<th>Direction</th>
<th>Top Rebar Arrangement</th>
<th>Bottom Rebar Arrangement</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Required Rebar (in²)</td>
<td>Provided Rebar</td>
</tr>
<tr>
<td>EDG slab at El. 100'-0&quot;</td>
<td>E-W (EDG-1)</td>
<td>0.43</td>
<td>#7 @ 9&quot;</td>
</tr>
<tr>
<td></td>
<td>N-S (EDG-1)</td>
<td>1.09</td>
<td>#9 @ 9&quot;</td>
</tr>
<tr>
<td></td>
<td>E-W (EDG-2)</td>
<td>1.04</td>
<td>#9 @ 9&quot;</td>
</tr>
<tr>
<td></td>
<td>N-S (EDG-2)</td>
<td>1.04</td>
<td>#9 @ 9&quot;</td>
</tr>
<tr>
<td>MSE slab at El. 137'-6&quot;</td>
<td>E-W</td>
<td>1.00</td>
<td>#9 @ 9&quot;</td>
</tr>
<tr>
<td></td>
<td>N-S</td>
<td>0.82</td>
<td>#10 @ 9&quot;</td>
</tr>
<tr>
<td>SFP slab at El. 114'-0&quot;</td>
<td>E-W</td>
<td>5.62</td>
<td>2-#14@9&quot;</td>
</tr>
<tr>
<td></td>
<td>N-S</td>
<td>5.99</td>
<td>2-#14@9&quot;</td>
</tr>
</tbody>
</table>

⁽¹⁾ Ratio = Provided Rebar / Required Rebar
Table 3.8A-34

Enveloped Design Forces of the EDG Building Basemat

<table>
<thead>
<tr>
<th>Element No.</th>
<th>Govern. Load Comb.</th>
<th>$N_x$ kip/ft</th>
<th>$N_y$ kip/ft</th>
<th>$M_x$ kip-ft/ft</th>
<th>$M_y$ kip-ft/ft</th>
<th>$V_{xz}$ kip/ft</th>
<th>$V_{yz}$ kip/ft</th>
<th>$V_{in, max}$ kip/ft</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>21020070</td>
<td>LC6</td>
<td>2.7</td>
<td>0.0</td>
<td>235.0</td>
<td>329.3</td>
<td>-97.9</td>
<td>-133.5</td>
<td>0.5</td>
<td></td>
</tr>
<tr>
<td>21020082</td>
<td>LC6</td>
<td>43.0</td>
<td>16.0</td>
<td>130.2</td>
<td>49.0</td>
<td>29.7</td>
<td>-15.2</td>
<td>1.2</td>
<td></td>
</tr>
<tr>
<td>21020109</td>
<td>LC6</td>
<td>-77.2</td>
<td>-83.4</td>
<td>358.8</td>
<td>946.9</td>
<td>-70.7</td>
<td>-95.3</td>
<td>37.8</td>
<td></td>
</tr>
<tr>
<td>21020138</td>
<td>LC6</td>
<td>17.9</td>
<td>55.6</td>
<td>100.0</td>
<td>294.2</td>
<td>-6.1</td>
<td>-3.9</td>
<td>18.7</td>
<td></td>
</tr>
<tr>
<td>21020186</td>
<td>LC6</td>
<td>-47.7</td>
<td>0.0</td>
<td>794.8</td>
<td>44.4</td>
<td>22.7</td>
<td>-11.0</td>
<td>7.2</td>
<td></td>
</tr>
<tr>
<td>21020188</td>
<td>LC6</td>
<td>-113.0</td>
<td>-40.5</td>
<td>503.9</td>
<td>489.1</td>
<td>133.9</td>
<td>-18.9</td>
<td>29.8</td>
<td></td>
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<tr>
<td>21020262</td>
<td>LC6</td>
<td>-76.4</td>
<td>-81.8</td>
<td>359.6</td>
<td>947.3</td>
<td>62.8</td>
<td>-55.6</td>
<td>37.3</td>
<td></td>
</tr>
<tr>
<td>21020264</td>
<td>LC6</td>
<td>-160.2</td>
<td>-13.6</td>
<td>604.4</td>
<td>113.7</td>
<td>-44.7</td>
<td>-7.0</td>
<td>14.2</td>
<td></td>
</tr>
</tbody>
</table>

(1) X is in EW direction, and Y is in NS direction.

(2) Design forces and moments in this table are the enveloped values of the following load combinations:

- LC4: 1.4D+1.7L
- LC6: 1.4D+1.7L+1.7W
- LC15a: 1.0D+1.0L+1.0Es
- LC15b: 0.9D+1.0Es
Table 3.8A-35

<table>
<thead>
<tr>
<th>Element Set No.</th>
<th>Elevation</th>
<th>Max./Min</th>
<th>N11 kip/ft</th>
<th>N22 kip/ft</th>
<th>M11 kip-ft/ft</th>
<th>M22 kip-ft/ft</th>
<th>Qout kip/ft</th>
<th>N12 kip/ft</th>
</tr>
</thead>
<tbody>
<tr>
<td>West wall of EDG building</td>
<td>100'-0&quot; to 135'-0&quot;</td>
<td>Max. 0.05</td>
<td>22.03</td>
<td>42.88</td>
<td>39.3</td>
<td>12.96</td>
<td>41.4</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Min -104.2</td>
<td>-20.7</td>
<td>-68.5</td>
<td>-32.8</td>
<td>-12.9</td>
<td>-41.3</td>
<td></td>
</tr>
<tr>
<td>Center wall of EDG building</td>
<td></td>
<td>Max. 0.0</td>
<td>21.19</td>
<td>8.56</td>
<td>3.77</td>
<td>5.67</td>
<td>38.94</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Min -68.59</td>
<td>-13.61</td>
<td>-8.68</td>
<td>-3.70</td>
<td>-5.68</td>
<td>-15.27</td>
<td></td>
</tr>
</tbody>
</table>

(1) Design forces and moments in this table are the enveloped values of the following load combinations:
- LC4: 1.4D+1.7L
- LC6: 1.4D+1.7L+1.7W
- LC15a: 1.0D+1.0L+1.0Es
- LC15b: 0.9D+1.0Es
## Table 3.8A-36

Required Reinforcement and Margins of Safety for the EDG Building Shear Wall

<table>
<thead>
<tr>
<th>Critical Section</th>
<th>Elevation</th>
<th>Horizontal</th>
<th>Vertical</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Required Rebar (in²)</td>
<td>Provided Rebar</td>
<td>Ratio (1)</td>
</tr>
<tr>
<td>West wall of EDG building</td>
<td>100'-0&quot; to 135'-0&quot;</td>
<td>1.171</td>
<td>#11@12&quot;</td>
</tr>
<tr>
<td>Center wall of EDG building</td>
<td>0.648</td>
<td>0.648</td>
<td>#11@12&quot;</td>
</tr>
</tbody>
</table>

(1) Ratio = Provided Rebar / Required Rebar
### Table 3.8A-37

Required Reinforcement and Margins of Safety for the EDG Building Basemat

<table>
<thead>
<tr>
<th>Location</th>
<th>EW-Direction</th>
<th>NS-Direction</th>
<th>Shear Bar</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Required Rebar (in^2)</td>
<td>Provided Rebar</td>
<td>Ratio(1)</td>
</tr>
<tr>
<td>EDG building at El.100'-0&quot;</td>
<td>2.074</td>
<td>2-#11@12&quot;</td>
<td>1.50</td>
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</table>

(1) Ratio = Provided Rebar / Required Rebar
Table 3.8A-38

Results on Factor of Safety for Basemat Stability

<table>
<thead>
<tr>
<th>Building</th>
<th>Item</th>
<th>FOS(1)</th>
<th>Allowable FOS</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>EDG</td>
<td>Overturning by Wind</td>
<td>10.67</td>
<td>1.5</td>
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</tr>
<tr>
<td></td>
<td>Overturning by Earthquake</td>
<td>1.58</td>
<td>1.1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sliding by Wind</td>
<td>5.41</td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sliding by Earthquake</td>
<td>1.82</td>
<td>1.1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Flotation</td>
<td>10.67</td>
<td>1.1</td>
<td></td>
</tr>
<tr>
<td>DFOT</td>
<td>Overturning by Wind</td>
<td>N/A</td>
<td></td>
<td>Buried structure</td>
</tr>
<tr>
<td></td>
<td>Overturning by Earthquake</td>
<td>1.19</td>
<td>1.1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sliding by Wind</td>
<td>N/A</td>
<td></td>
<td>Buried structure</td>
</tr>
<tr>
<td></td>
<td>Sliding by Earthquake</td>
<td>1.29</td>
<td>1.1</td>
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<tr>
<td></td>
<td>Flotation</td>
<td>1.81</td>
<td>1.1</td>
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(1)  FOS = Factor of Safety
Table 3.8A-39

EDG & DFOT buildings Differential Settlements According to Site Profiles (Static)

<table>
<thead>
<tr>
<th>Location</th>
<th>Node #1</th>
<th>Node #2</th>
<th>Distance (ft)</th>
<th>Differential Settlement (inches)</th>
<th>Soil #1</th>
<th>Soil #4</th>
<th>Soil #8</th>
</tr>
</thead>
<tbody>
<tr>
<td>EDG</td>
<td>4451</td>
<td>4036</td>
<td>41.291</td>
<td>0.181</td>
<td>0.103</td>
<td>0.043</td>
<td></td>
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<tr>
<td></td>
<td>4036</td>
<td>4774</td>
<td>36.867</td>
<td>0.024</td>
<td>0.033</td>
<td>0.030</td>
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<tr>
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<td>4036</td>
<td>131</td>
<td>47.734</td>
<td>0.155</td>
<td>0.077</td>
<td>0.029</td>
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</tr>
<tr>
<td></td>
<td>4036</td>
<td>97</td>
<td>47.734</td>
<td>0.076</td>
<td>0.002</td>
<td>0.016</td>
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</tr>
<tr>
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<td>131</td>
<td>8308</td>
<td>47.734</td>
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<td>0.076</td>
<td>0.029</td>
<td></td>
</tr>
<tr>
<td></td>
<td>8308</td>
<td>97</td>
<td>47.734</td>
<td>0.080</td>
<td>0.003</td>
<td>0.016</td>
<td></td>
</tr>
<tr>
<td></td>
<td>8308</td>
<td>8678</td>
<td>41.291</td>
<td>0.182</td>
<td>0.103</td>
<td>0.043</td>
<td></td>
</tr>
<tr>
<td></td>
<td>8308</td>
<td>8953</td>
<td>41.291</td>
<td>0.045</td>
<td>0.027</td>
<td>0.031</td>
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<tr>
<td></td>
<td>4460</td>
<td>8923</td>
<td>33.253</td>
<td>0.032</td>
<td>0.006</td>
<td>0.002</td>
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<tr>
<td></td>
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<td></td>
<td></td>
<td>Total Max. Differential Settlement</td>
<td>0.182</td>
<td>0.103</td>
<td>0.043</td>
</tr>
<tr>
<td>DFOT</td>
<td>5876</td>
<td>794</td>
<td>45.881</td>
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<td>7068</td>
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<td>304</td>
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<td>0.005</td>
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<td>7059</td>
<td>25.836</td>
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<td>26.023</td>
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<td>0.011</td>
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<td>63</td>
<td>5872</td>
<td>26.023</td>
<td>0.323</td>
<td>0.080</td>
<td>0.006</td>
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<td>6604</td>
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<td>14.916</td>
<td>0.194</td>
<td>0.061</td>
<td>0.005</td>
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<tr>
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<td></td>
<td></td>
<td></td>
<td>Total Max. Differential Settlement</td>
<td>0.384</td>
<td>0.140</td>
<td>0.025</td>
</tr>
</tbody>
</table>
### Table 3.8A-40 (1 of 2)

#### Summary of Models and Analysis Methods

<table>
<thead>
<tr>
<th>Model</th>
<th>Analysis Method</th>
<th>Program</th>
<th>Purpose</th>
<th>Subsections</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor containment building analysis model (Uncracked stiffness model)</td>
<td>(2) Modal analysis (3) Response spectrum analysis (SSE damping) (4) Static analysis (5) Heat transfer analysis</td>
<td>ANSYS</td>
<td>To generate the design forces of the reactor containment building shell and dome (e.g. dead, live, seismic thermal load, etc.)</td>
<td>Subsection 3.8A.1.4.1</td>
</tr>
<tr>
<td>Reactor containment building internal structure model (Uncracked stiffness model)</td>
<td>(6) Modal analysis (7) Response spectrum analysis (SSE damping) (8) Static analysis</td>
<td>ANSYS</td>
<td>To generate the design forces of the reactor containment building internal structure walls (e.g. dead, live, seismic thermal load, etc.)</td>
<td>Subsection 3.8A.1.4.3</td>
</tr>
<tr>
<td>Reactor containment building - IRWST hydro-dynamic analysis model (Uncracked stiffness model)</td>
<td>(9) Direct integration time history analysis</td>
<td>ANSYS</td>
<td>To generate the floor response spectrum due to the POSRV sparger discharge load</td>
<td>Subsection 3.8A.1.4.3.1.3</td>
</tr>
<tr>
<td>NI building common basemat analysis model (Uncracked stiffness model)</td>
<td>(10) Nonlinear Analysis</td>
<td>ANSYS</td>
<td>To generate the design forces of the NI common basemat (e.g. dead, live, seismic load, etc.)</td>
<td>Subsection 3.8A.1.4.2, 3.8A.2.4.1</td>
</tr>
<tr>
<td>Auxiliary building structural analysis model</td>
<td>(11) Static analysis (Equivalent static method)</td>
<td>ANSYS</td>
<td>To generate the design forces of the auxiliary building shear walls (e.g. dead, live, seismic load, etc.)</td>
<td>Subsection 3.8A.2.3, 3.8A.2.4.2</td>
</tr>
<tr>
<td>Auxiliary building - SFP hydro-dynamic analysis model (Uncracked stiffness model)</td>
<td>(12) Modal analysis (13) Static analysis</td>
<td>ANSYS</td>
<td>To create the auxiliary building SFP hydrodynamic force</td>
<td>Subsection 3.8A.2.4.2</td>
</tr>
<tr>
<td>Auxiliary building - SFP, local analysis model (Uncracked stiffness model)</td>
<td>(14) Heat transfer analysis (15) Static analysis</td>
<td>ANSYS</td>
<td>To generate the design forces of the auxiliary building SFP walls</td>
<td>Subsection 3.8A.2.4.2</td>
</tr>
<tr>
<td>Auxiliary building - aux. feed water storage tank hydro-dynamic analysis model (Uncracked stiffness model)</td>
<td>(16) Modal analysis (17) Static analysis</td>
<td>ANSYS</td>
<td>To create the auxiliary building AFWT hydrodynamic force</td>
<td>Subsection 3.8A.2.4.2</td>
</tr>
<tr>
<td>Model</td>
<td>Analysis Method</td>
<td>Program</td>
<td>Purpose</td>
<td>Subsections</td>
</tr>
<tr>
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<td>------------------------------------------------------</td>
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<td>---------------------------------------------------------------------------------------------</td>
<td>------------------------------</td>
</tr>
<tr>
<td>Emergency diesel generator building structural analysis model (Uncracked stiffness model)</td>
<td>(18) Static analysis (Equivalent static method)</td>
<td>ANSYS</td>
<td>To generate the design forces of the emergency diesel generator building shear wall and basemat (e.g. dead load, live load, seismic load, etc.)</td>
<td>Subsection 3.8A.3.4.1, 3.8A.3.4.2</td>
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<td>(19) Nonlinear Analysis</td>
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<td></td>
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<tr>
<td>Diesel fuel oil storage tank building structure analysis model (Uncracked stiffness model)</td>
<td>(20) Static analysis (Equivalent static method)</td>
<td>ANSYS</td>
<td>To generate the design forces of the diesel fuel oil storage tank building shear wall and basemat (e.g. dead load, live load, seismic load, etc.)</td>
<td>Subsection 3.8A.3.4.1, 3.8A.3.4.2</td>
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<tr>
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<td>(21) Nonlinear analysis</td>
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<td></td>
<td></td>
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<tr>
<td>Reactor containment building – combustible gas control inside containment analysis model (Uncracked stiffness model)</td>
<td>(22) Nonlinear analysis</td>
<td>ABAQUS</td>
<td>To evaluate the structural integrity of the reactor containment building under severe accident pressure in accordance with ASME CC-3720</td>
<td>Subsection 3.8.1.4.12</td>
</tr>
<tr>
<td>Reactor containment building - ultimate pressure capacity analysis model</td>
<td>(23) Nonlinear analysis</td>
<td>ABAQUS</td>
<td>To evaluate the ultimate pressure capacity of the reactor containment building</td>
<td>Subsection 3.8.1.4.11</td>
</tr>
<tr>
<td>Reactor containment building - reinforced concrete section model</td>
<td>(24) Static analysis</td>
<td>DARTEM</td>
<td>To calculate stress and strain of reinforced concrete sections under mechanical and temperature loads</td>
<td>Subsection 3.8A.1.4.1.3.7, 3.8A.1.4.2.3</td>
</tr>
<tr>
<td>Auxiliary building - SFP reinforced concrete section model</td>
<td>(25) Static analysis</td>
<td>DARTEM</td>
<td>To calculate stress and strain of reinforced concrete sections under mechanical and temperature loads</td>
<td>Subsection 3.8A.2.4.2, 3.8A.2.4.3</td>
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<tr>
<td>Reactor containment building - liner plate anchorage system model</td>
<td>(26) Static analysis</td>
<td>LBAP</td>
<td>To calculate maximum anchor forces and displacements in the liner anchorage systems attached to concrete walls</td>
<td>Subsection 3.8.1.4.10</td>
</tr>
<tr>
<td>Auxiliary building – concrete slab analysis model</td>
<td>(27) Static analysis</td>
<td>GTSTRUDL</td>
<td>To generate the design forces for the concrete slabs of the auxiliary building</td>
<td>Subsection 3.8A.2.4.3</td>
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</tbody>
</table>
# Table 3.8A-41

**Stress Profiles of Typical Prestressing Tendons**

## Vertical Tendon

<table>
<thead>
<tr>
<th>Stress Point</th>
<th>Stress in Tendon, MPa (ksi)</th>
<th>Losses of Stress, MPa (ksi)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Initial (1)</td>
<td>Final (2)</td>
</tr>
<tr>
<td>Anchor Point</td>
<td>1291.0 (187.24)</td>
<td>1007.0 (146.06)</td>
</tr>
<tr>
<td>Spring Line</td>
<td>1372.1 (199.00)</td>
<td>1088.1 (157.81)</td>
</tr>
<tr>
<td>Dome Apex</td>
<td>1089.9 (158.08)</td>
<td>806.0 (116.90)</td>
</tr>
</tbody>
</table>

Total Loss: 284.0 (41.18)

## Horizontal Tendon

<table>
<thead>
<tr>
<th>Stress Point</th>
<th>Stress in Tendon, MPa (ksi)</th>
<th>Losses of Stress, MPa (ksi)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Initial (1)</td>
<td>Final (2)</td>
</tr>
<tr>
<td>Anchor Point</td>
<td>1291.0 (187.24)</td>
<td>914.2 (132.60)</td>
</tr>
<tr>
<td>Tangent Point</td>
<td>1300.1 (188.57)</td>
<td>923.4 (133.93)</td>
</tr>
<tr>
<td>Midpoint of Tendon</td>
<td>1080.1 (156.66)</td>
<td>703.3 (102.01)</td>
</tr>
</tbody>
</table>

Total Loss: 376.8 (54.65)

---

(1) At time of tendon lock off  
(2) 60 years after plant startup
### Table 3.8A-42

**Margins of Safety for In-containment Refueling Water Storage Tank**

<table>
<thead>
<tr>
<th>Location</th>
<th>Meridional/Vertical Direction</th>
<th>Hoop/Horizontal Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Reinforcement</td>
<td>Required (in²)</td>
</tr>
<tr>
<td>Top Slab</td>
<td>1.16</td>
<td>#11@ 0.9&quot;</td>
</tr>
<tr>
<td>Outer Wall</td>
<td>0.073</td>
<td>#11@ 0.9&quot;</td>
</tr>
</tbody>
</table>

(1) Ratio = Provided Rebar / Required Rebar
Table 3.8A-43

Design Forces and Moments for Slab in RCB

<table>
<thead>
<tr>
<th>Location</th>
<th>Thick -ness</th>
<th>Directi on</th>
<th>Design Force and Moments</th>
<th>N_{xx} (kip/ft)</th>
<th>N_{yy} (kip/ft)</th>
<th>N_{xy} (kip/ft)</th>
<th>M_{xx} (kip-ft/ft)</th>
<th>M_{yy} (kip-ft/ft)</th>
<th>M_{xy} (kip-ft/ft)</th>
<th>V_{out} (kip/ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2ft</td>
<td>Radial</td>
<td>Operating Floor Slab at El. 156'-0&quot;</td>
<td>Area (1)</td>
<td>165.56</td>
<td>-</td>
<td>8.24</td>
<td>145.31</td>
<td>-</td>
<td>-1.55</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Area (2)</td>
<td>35.75</td>
<td>-</td>
<td>7.58</td>
<td>88.44</td>
<td>-</td>
<td>-3.71</td>
<td>34.31</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Tangential</td>
<td>All Areas</td>
<td>-</td>
<td>3.67</td>
<td>-65.54</td>
<td>-</td>
<td>9.58</td>
<td>-49.27</td>
<td></td>
</tr>
<tr>
<td>3ft</td>
<td>Radial</td>
<td>at SSW Area (1)</td>
<td>53.93</td>
<td>-</td>
<td>3.60</td>
<td>287.49</td>
<td>-</td>
<td>-14.96</td>
<td>28.43</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>at Central Area (2)</td>
<td>38.72</td>
<td>-</td>
<td>2.92</td>
<td>192.15</td>
<td>-</td>
<td>-19.05</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Tangential</td>
<td>All Areas</td>
<td>-</td>
<td>-64.97</td>
<td>10.34</td>
<td>-</td>
<td>-65.69</td>
<td>30.36</td>
<td></td>
</tr>
</tbody>
</table>

(1) These forces are considered when designing the rebar for the connection area between slab and SSW.
(2) These forces are considered when designing the rebar for the central area of slab.
Table 3.8A-44

Slab Reinforcement and Margins of Safety at Each Critical Section in RCB

<table>
<thead>
<tr>
<th>Location</th>
<th>Thickness</th>
<th>Radial Direction</th>
<th>Tangential Direction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Area</td>
<td>Required Rebar (in²)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Area</td>
<td></td>
</tr>
<tr>
<td>Operating Floor slab at El. 156'-0&quot;</td>
<td>2ft</td>
<td>at SSW Area</td>
<td>3.81</td>
</tr>
<tr>
<td></td>
<td></td>
<td>at Central Area</td>
<td>1.75</td>
</tr>
<tr>
<td></td>
<td>3ft</td>
<td>at SSW Area</td>
<td>3.05</td>
</tr>
<tr>
<td></td>
<td></td>
<td>at Central Area</td>
<td>2.12</td>
</tr>
</tbody>
</table>

(1) Ratio = Provided Rebar / Required Rebar
## Table 3.8A-45

### Liner Plate Allowables

<table>
<thead>
<tr>
<th>Category</th>
<th>Stress-Strain Allowable</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Membrane</td>
</tr>
<tr>
<td>Construction</td>
<td>$f_{st} = f_{sc} = \frac{2}{3} f_{py}$</td>
</tr>
<tr>
<td>Service</td>
<td>$\varepsilon_{st} = \varepsilon_{sc} = 0.002$ in/in</td>
</tr>
<tr>
<td>Factored</td>
<td>$\varepsilon_{sc} = 0.005$ in/in</td>
</tr>
<tr>
<td></td>
<td>$\varepsilon_{st} = 0.003$ in/in</td>
</tr>
</tbody>
</table>
### Table 3.8A-46

**Liner Anchor Allowables**

<table>
<thead>
<tr>
<th>Category</th>
<th>Force and Displacement Allowables</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mechanical Loads, Losser of:</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>Test</td>
<td></td>
</tr>
<tr>
<td>Normal</td>
<td>$F_a = 0.67 , F_y$</td>
</tr>
<tr>
<td>Severe Environmental</td>
<td>$F_a = 0.33 , F_u$</td>
</tr>
<tr>
<td>Extreme Environmental</td>
<td></td>
</tr>
<tr>
<td>Abnormal</td>
<td>$F_a = 0.90 , F_y$</td>
</tr>
<tr>
<td>Abnormal/Severe Environmental</td>
<td>$F_a = 0.50 , F_u$</td>
</tr>
<tr>
<td>Abnormal/Extreme Environmental</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$\delta_a = 0.25 , \delta_u$</td>
</tr>
<tr>
<td></td>
<td>$\delta_a = 0.50 , \delta_u$</td>
</tr>
</tbody>
</table>
### Table 3.8A-47

#### Design Result of Steel Beams and Connections

<table>
<thead>
<tr>
<th>Elevation</th>
<th>Beam Size</th>
<th>Member Stress Check Interaction Ratio (IR)</th>
<th>Additional Stress Check due to Torsion</th>
<th>IR Friction</th>
<th>Total IR</th>
</tr>
</thead>
<tbody>
<tr>
<td>114'-0&quot;</td>
<td>Below 2ft Concrete Slab W21x147</td>
<td>0.563</td>
<td>0.086</td>
<td>0.039</td>
<td>0.688</td>
</tr>
<tr>
<td>136'-6&quot;</td>
<td>Below 3ft Concrete Slab W30x261</td>
<td>0.307</td>
<td>0.036</td>
<td>0.025</td>
<td>0.418</td>
</tr>
<tr>
<td>156'-0&quot;</td>
<td>Below 3ft Concrete Slab BW24x270</td>
<td>0.405</td>
<td>0.026</td>
<td>0.025</td>
<td>0.516</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Connection Type</th>
<th>Maximum Load (kips)</th>
<th>Allowable Load (kips)</th>
<th>Ratio(^{(1)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>Beam Seat Plate at SSW</td>
<td>98.1</td>
<td>182.3</td>
<td>0.53</td>
</tr>
<tr>
<td>Beam Seat Plate at Containment Wall</td>
<td>100.5</td>
<td>218.8</td>
<td>0.45</td>
</tr>
<tr>
<td>Beam Seat Welding at SSW</td>
<td>300.3</td>
<td>528.9</td>
<td>0.56</td>
</tr>
<tr>
<td>Beam Seat Welding at Containment Wall</td>
<td>307.9</td>
<td>634.7</td>
<td>0.48</td>
</tr>
<tr>
<td>Lower Key Bumper</td>
<td>20.6</td>
<td>60.0</td>
<td>0.34</td>
</tr>
<tr>
<td>Welding at SSW</td>
<td>100.0</td>
<td>103.0</td>
<td>0.97</td>
</tr>
</tbody>
</table>

\(^{(1)}\) Ratio = Maximum Load / Allowable Load
Figure 3.8A-1  Schematic View of Equipment Hatch and Personnel Airlocks
Security-Related Information – Withhold Under 10 CFR 2.390
Figure 3.8A-3  Tendon Model Using Truss Elements

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Figure 3.8A-4  FE Partial Model of Containment

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(a) ISRS at Internal Structure El 78'-0" (3% Damping)

(b) ISRS at Containment Wall El 78'-0" (5% Damping)

(c) ISRS at Internal Structure El 78'-0" (7% Damping)

Figure 3.8A-5  Input Acceleration Response Spectrum at El. 78'-0" of RCB
Figure 3.8A-6  Rebar Arrangement of Wall-Basemat Junction Area
Figure 3.8A-7  Rebar Arrangement of Wall Mid-Height
Figure 3.8A-8  Rebar Arrangement of Polar Crane Bracket Level and Springline

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Figure 3.8A-9  Rebar Arrangement of Wall around Equipment Hatch

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Figure 3.8A-10  Rebar Arrangement of Wall around Personnel Airlock
Figure 3.8A-12  Full FE Model for the Basemat Structural Analysis
Figure 3.8A-13 Model for AB, RCB Wall, Dome, Internal Structure, and Basemat

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Figure 3.8A-14  LINK180 Element for Soil Spring
Figure 3.8A-15 Design Sections for Basemat Reinforcement

3.8A-119
Figure 3.8A-16  Reinforcement Arrangement of RCB Basemat (East-West View)
Figure 3.8A-17 Reinforcement Arrangement of RCB Basemat (South-North View)
Figure 3.8A-18  Node Location at NI Basemat for Checking Settlement (Static)
Figure 3.8A-19  Full FE Model for the RCB Internal Structure
(a) Solid Element Model (PSW, IRWST, and Fill Concrete)

(b) Shell Element Model (SSW)

(c) Beam Element Model (RCS)

Figure 3.8A-20  Solid, Shell, and Beam Element Model for RCB Internal Structure
Figure 3.8A-21 Location of Critical Sections of the AB at El. 100′-0″
Figure 3.8A-22 Location of Critical Sections of the AB at El. 120’-0"
Figure 3.8A-23  Location of Critical Sections of the AB at El. 137′-6″

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Figure 3.8A-24  Location of Critical Sections of the AB at El. 156′-0″

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Figure 3.8A-25  Solid Element Model of NI Common Basemat
Figure 3.8A-26  3-D View of FE Model for AB Superstructure
Figure 3.8A-27  Components of FE Model for Rectangular AFW Tank in AB
Figure 3.8A-28  Partial FE Model for FHA Pools in AB
Figure 3.8A-29 FE Model for AB Slab Design at El. 157'6"
Figure 3.8A-30  Full FE Model for the EDG Building
Figure 3.8A-31  FE Model for the EDG Building Basemat
Figure 3.8A-32 Reinforcement Arrangement of the AB Basemat
Figure 3.8A-33  Typical Reinforcement of the AB Basemat (Section A)
Figure 3.8A-34  Typical Reinforcement of the AB Basemat (Section B)
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Figure 3.8A-35 Typical Reinforcement Details of Basemat

Note: $L_{dT}$ - development length for top horizontal bar
(reinforcement location factor = 1.3, ACI 349 12.2.4)
$L_{dT}$ - lap splice length for top horizontal bar
$L_{dO}$ - development length for other bars
$L_{bO}$ - lap splice length for other bars
Figure 3.8A-36  Reinforcement Arrangement of the AB MSIV House Wall (Section 1)
Figure 3.8A-37  Reinforcement Arrangement of the AB MSIV House Wall (Section 2)
Figure 3.8A-38  Reinforcement Arrangement of the AB AFWST Wall
Figure 3.8A-39  Reinforcement Arrangement of the AB MCR Wall (Section 1)
Figure 3.8A-40  Reinforcement Arrangement of the AB MCR Wall
(Section 2)
Figure 3.8A-41  Reinforcement Arrangement of the AB SFP Wall
Figure 3.8A-42 Reinforcement Arrangement of the AB FHA Wall (Section 1)
Figure 3.8A-43 Reinforcement Arrangement of the AB FHA Wall (Section 2)
Note: $L_{dt}$ - development length for top horizontal bar
(reinforcement location factor = 1.3, ACI 349 12.2.4)

$L_{dt}$ - lap splice length for top horizontal bar
$L_{do}$ - development length for other bars
$L_{bo}$ - lap splice length for other bars

Figure 3.8A-44  Typical Shear Wall Reinforcement Arrangement Detail
Figure 3.8A-45 Slab Reinforcements for the EDG Room in AB
Figure 3.8A-46 Slab Reinforcements for the SFP in AB
Figure 3.8A-47  Slab Reinforcements for the MSE in AB
Note: \( L_{dT} \) - development length for top horizontal bar
(reinforcement location factor = 1.3, ACI 349 12.2.4)
\( L_{ST} \) - lap splice length for top horizontal bar
\( L_{dO} \) - development length for other bars
\( L_{SO} \) - lap splice length for other bars

Figure 3.8A-48 Typical Slab Reinforcement Arrangement Data
Security-Related Information – Withhold Under 10 CFR 2.390

Figure 3.8A-49 Location of Critical Sections of the EDG Building
Figure 3.8A-50 Typical Reinforcement of the EDG Building Basemat
Figure 3.8A-51 Reinforcement Arrangement of Center Wall of the EDG Building
Figure 3.8A-52 Reinforcement Arrangement of the West Wall of the EDG Building
Figure 3.8A-53  Rebar Arrangement of Containment Dome
Figure 3.8A-54 Node Locations at EDG Basemat for Checking Settlements (Static)
Figure 3.8A-55 Node Locations at DFOT Basemat for Checking Settlements (Static)
Figure 3.8A-56  Concrete Slab and Steel Beam between Containment Wall and SSW
Figure 3.8A-57 Slab Reinforcements for Operating Floor at El. 156'-0" in RCB
Figure 3.8A-58  Reinforcement Arrangement for Primary Shield Wall
Figure 3.8A-59  Reinforcement Arrangement for Secondary Shield Wall
Figure 3.8A-60  Reinforcement Arrangement for Refueling Pool Wall
Figure 3.8A-61 Reinforcement Arrangement for SG Enclosure Wall
Figure 3.8A-62  Reinforcement Arrangement for PZR Enclosure Wall
Figure 3.8A-63  Emergency Diesel Generator Building Block Section
Figure 3.8A-64  Emergency Diesel Generator Building Block Plan
3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The following information identifies the transients used in the design and fatigue analysis of ASME Code Class 1 components, reactor internals, and component supports. All transients are classified with respect to the component operating conditions identified as Level A (Normal), B (Upset), C (Emergency), D (Faulted), and testing as defined in the ASME Section III. The transients specified below represent conservative estimates for design purposes only and are not intended to represent actual transients, nor necessarily reflect actual operating procedures; nevertheless, all envisaged actual transients are accounted for, and the number and severity of the design transients exceeds those that may be anticipated during the life of the plant.

Pressure, temperature, and flow rate resulting from the normal, test, upset, emergency, and faulted transients are computed by means of computer simulations of the nuclear steam supply system (NSSS) components. Design transients are detailed in the design specifications via time-history plots of the fluid temperature, pressure, and flow rate during plant events. The component designer then uses the transient data in the design specification as the basis for design and fatigue analysis. In support of the design of each Code Class 1 and core support (CS) component, a fatigue analysis for the combined effects of mechanical and thermal loads is performed in accordance with the requirements of ASME Section III. The purpose of the analysis is to demonstrate that fatigue failure will not occur when the components are subjected to typical dynamic events that may occur during the life of the plant.

ASME Section III, Division 1, Subsection NB, and NRC RG 1.207 are used for performing fatigue evaluations considering the effects of reactor coolant environment of the APR1400 components.

The fatigue analysis is based on a series of dynamic events depicted in the respective design specifications. Associated with each dynamic event is a mechanical, thermal-hydraulic transient presentation along with an assumed number of occurrences for the event. The presentation is generally simple and straightforward because it is meant to envelop the actual plant responses. The intent is to present material for purposes of design.
APR1400 DCD TIER 2

The fundamental concept provides reasonable assurance that the consequences of the normal and upset conditions that are expected to occur in the power plant are enveloped by one or more of the dynamic event portrayals in the design specifications. The number of occurrences selected for each dynamic event is conservative so that in the aggregate, a 60-year useful life is provided by the design process.

Design loading combinations for ASME Code Class 1, 2, and 3 components are given in Subsection 3.9.3. Design loading combinations for reactor internals structures are presented in Subsection 3.9.5.2.

The principal design bases of the RCS and reactor internals structures are given in Section 5.2 and Subsection 3.9.5, respectively.

The APR1400 design basis initiating events and frequencies used in the stress analysis of ASME Code Class 1 and Class CS components of the primary system are shown in Table 3.9-1. The resulting APR1400 events and frequencies conservatively represent the 60-year design basis.

The design basis events (DBEs) are classified as normal, upset, emergency, faulted, and test. The normal and test events are planned operations that occur during the life of the plant. Upset events are occurrences that may occur during the life of the plant. Emergency and faulted events are not expected to occur but are included in the design basis for additional design margin. The normal and test events are selected by reviewing the expected plant operations. The upset, emergency, and faulted events are determined by reviewing industry databases (References 1, 2, 3, and 4) for events that have occurred, or that may be postulated to occur, based on observed plant behavior.

Normal and test event frequencies are determined by summing the number of expected plant operations over the 60-year design life. The frequencies for upset, emergency, and faulted events are determined on a probabilistic basis using industry databases (References 1, 2, 3, and 4). The 60-year design frequency of occurrence stated in Table 3.9-1 is always greater than the expected frequency of occurrence.

Conservative mathematical models and methodology are used to determine the thermal-hydraulic consequences of the DBEs on individual plant components. The design margin is further enhanced by enveloping similar events and using the most conservative thermal-hydraulic consequences to represent a composite group. The group frequency is then determined by algebraically summing the individual design frequencies.
Pressure and thermal stress variations associated with the design transients are considered in the design of supports, valves, and piping within the reactor coolant pressure boundary (RCPB).

In addition to the design transients listed above and included in the fatigue analysis, the loadings produced by seismic events are also applied in the design of components and support structures of the RCS. The number of cycles pertaining to fatigue effects of cyclic motion associated with the seismic events is provided in Subsection 3.7.3. Design loading combinations for ASME Code Class 1, 2, and 3 components are addressed in Subsection 3.9.3.

ASME Section III defines the plant conditions (Service Level A, B, C, and D, and Test Conditions) for the design of RCS Class 1 components, auxiliary Class 1 components, RCS component supports, and reactor internals as described below.

Normal Conditions (ASME Service Level A)

Normal conditions include any condition in the course of startup, operation in the design power range, hot standby, and system shutdown other than upset, emergency, faulted, or testing conditions.

Upset Conditions (Incidents of Moderate Frequency; ASME Service Level B)

Upset conditions include any deviations from normal conditions that are anticipated to occur often enough for the design to include a capability to withstand the conditions without operational impairment. Upset conditions include transients that result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of power. Upset conditions also include abnormal incidents not resulting in a forced outage as well as those that cause forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition is included in the design specifications.

Emergency Conditions (Infrequent Incidents; ASME Service Level C)

Emergency conditions include deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the system. The emergency conditions have a low probability of occurrence, but are included to demonstrate that no gross loss of
structural integrity will result as a concomitant effect of any damage developed in the system.

Faulted Conditions (ASME Service Level D)

Faulted conditions are combinations of conditions associated with low probability, postulated events whose consequences may impair the integrity and operability of the nuclear energy system to the extent that consideration of public health and safety are involved. Such considerations require conformance with safety criteria. The methods of analysis to calculate the stresses and deformations conform with the methods in ASME Section III, Division 1, Appendix F.

Testing Conditions

Testing conditions include hydrostatic pressure tests of individual components and the primary system as specified in this section.

In accordance with ASME Section III, emergency and faulted conditions are not included in fatigue evaluations, with the exception that any significant emergency cycles in excess of 25 are considered in the fatigue analyses.

3.9.1.1.1 Service Level A Conditions

Service Level A conditions consist of 14 event conditions as shown in Table 3.9-1. Each event condition may include several specific events based on the thermal-hydraulic characteristics of the event. The following are the event descriptions for specific events considered in Service Level A conditions.

a. Steady-state operation with normal NSSS parameter variations in the increasing and decreasing directions.

The plant could undergo primary and secondary process parameter variations because of secondary steam conditions. Power step changes of 10 percent are used to envelop this event. This is conservative since the 10 percent power step change produces more severe plant process parameter variations than those that occurred during any normal plant variations as the result of changing steam conditions.
Each event is assumed to occur 1,500,000 times during the 60-year plant design life.

b. Daily load follow operation

Although APR1400 will be operated as a base load plant, the effects of daily load follow operation are accounted for in the structural design and analysis of ASME Code Class 1 components, reactor internals, and component supports.

For daily load follow operation, the power is maintained at 100 percent for 10 through 16 hours, ramped down from 100 percent to 50 percent over a 2-hour period, operated for 4 through 10 hours at 50 percent, and then ramped up from 50 percent to 100 percent power over a 2-hour period.

Each event is assumed to occur 22,000 times during the 60-year plant design life.

c. Turbine power step changes of 10 percent power (15 to 100 percent power)

This event is a turbine power change from 100 to 90 percent power and from 90 to 100 percent power. The transients of step change from 25 to 15 percent power and step change from 15 to 25 percent power are also considered. These power step changes are representative of other power levels and serve to envelop smaller power step changes.

Each event is assumed to occur 3,200 times during the 60-year plant design life.

d. Turbine power step changes of 1 percent power (5-15 percent)

This power step change is from 15 to 14 percent power and from 14 to 15 percent power. The transients of step change from 6 to 5 percent power and step change from 5 to 6 percent power are also considered. These power steps are representative of other low power levels and serve to envelop other possible power steps.

Each event is assumed to occur 1,600 times during the 60-year plant design life.

e. Turbine load rejection up to 50 percent power (50-100 percent power)
This event is a load rejection up to 50 percent power. The load rejection from 100 percent to 50 percent power is more severe than any other smaller load rejection.

This event is assumed to occur 60 times during the 60-year plant design life.

f. Turbine generator runback to house load

This event is a loss of offsite load with the turbine running back to house load. The house load is about 5 percent of full-power conditions.

This event is assumed to occur 60 times during the 60-year plant design life.

g. Reactor trip

This event is an uncomplicated reactor trip. The uncomplicated reactor trip is an event when the reactor trip is the event initiator. A reactor trip can occur at any power level and causes a turbine trip.

This event is assumed to occur 150 times during the 60-year plant design life.

h. Turbine trip

This event is a turbine trip caused by a mechanical or electrical problem.

This event is assumed to occur 150 times during the 60-year plant design life.

i. Turbine power ramp changes of 5 percent/min (15-100 percent power)

This event is a turbine power ramp change from 100 to 15 percent power and power ramp change from 15 to 100 percent power. The turbine power ramp changes of 5 percent/min between these power levels are more severe than power ramps from any other power levels and serve to envelop the less severe power ramps.

Each event is assumed to occur 3,200 times during the 60-year plant design life.

j. Turbine power ramp changes of 1 percent /min (5-15 percent power)
This event is a turbine power ramp change from 15 percent to 5 percent power and power ramp change from 5 percent to 15 percent power. The turbine power ramp changes of 1 percent/min between these power levels are more severe than power ramps from any other power levels and serve to envelop the less severe power ramps.

Each event is assumed to occur 1,600 times during the 60-year plant design life.

k. Loss of main FW pumps without reactor trip

The loss of main feedwater pumps without generating a reactor trip event is composed of two events: loss of a main feedwater pump and loss of two main feedwater pumps. The loss of one main feedwater pump results in a minor system transient. The two remaining feedwater pumps automatically increase their speeds to match the flow rate requirements of the original power level. The loss of two main feedwater pumps at 100 percent power envelops all other possible cases that may occur during part load operation.

This event is assumed to occur 60 times during the 60-year plant design life.

l. NSSS operations with manual control of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control, and feedwater flow (0 to 5 percent power)

This event consists of several manual operations that can be expected to occur during low-power conditions. These include the manual operation of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control, and the feedwater flow control.

This event is assumed to occur 1,600 times during the 60-year plant design life.

m. Opening or closure of the economizer feedwater control valve

The steam generator has two feedwater control valves to provide the necessary flow control over the full power range. The smaller downcomer valve controls flow between 0 percent and 20 percent power and the larger economizer feedwater valve controls flow between 20 percent and 100 percent power. The feedwater valve switch is performed at 20 percent reactor power during power increase and at 18 percent reactor power during power decrease.
This event is assumed to occur 500 times during the 60-year plant design life.

n. NSSS operations with the NSSS control systems in the manual mode (5-100 percent power)

This event consists of several manual operations that can be expected to occur during the 5 to 100 percent power range. It includes manual operation of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control, and the feedwater flow control. The control systems can be manually controlled within the normal automatic control bands.

This event is assumed to occur 3,200 times during the 60-year plant design life.

o. Manual operation of the auxiliary spray system

The manual operation of the auxiliary spray system may be required during power operation to reduce primary pressure excursions when the main spray system is out-of-service. This transient may be especially severe for the pressurizer spray line since isolated fluid in the auxiliary spray line may cool down to low temperatures before being sprayed into the pressurizer.

This event is assumed to occur 250 times during the 60-year plant design life.

p. High-capacity steam generator blowdown

The steam generator high capacity blowdown is performed with a flow rate of approximately 5 percent of the steam generator maximum steaming rate to maintain steam generator chemistry within control limit.

This event is assumed to occur 3,200 times during the 60-year plant design life.

q. Shift from normal to maximum CVCS flow rate

The chemical and volume control system (CVCS) letdown and charging flow rates may be increased to support required changes in boron concentration or to more rapidly reduce impurities in the RCS.

This event is assumed to occur 3,200 times during the 60-year plant design life.

r. Low-low VCT level and charging pump diversion to the boric acid storage tank
Low volume control tank (VCT) level results in diverting the charging flow sources from VCT to the boric acid storage tank.

This event is assumed to occur 60 times during the 60-year plant design life.

s. Spurious actuation of the pressurizer spray

The spurious actuation of the pressurizer spray may occur because of a mechanical or a control system failure. The failure is assumed to open both spray valves while the plant is operating at full power. This is the more severe event of the possible spray valve failures and envelops other possible failures.

This event is assumed to occur 60 times during the 60-year plant design life.

t. Spurious actuation of the pressurizer heaters

The spurious actuation of the pressurizer heaters may occur because of a mechanical or a control system failure. The failure is assumed to actuate all pressurizer heaters while the plant is operating at full power and at hot standby.

This event is assumed to occur 60 times during the 60-year plant design life.

u. Inadvertent closure of one economizer or downcomer FW control valve

The spurious closure of one economizer or downcomer feedwater control valve may occur because of a mechanical or a control system failure. The failure will result in the loss of feedwater flow to the economizer section of one steam generator.

This event is assumed to occur 60 times during the 60-year plant design life.

v. Inadvertent opening of one economizer or downcomer FW control valve

The spurious opening of one economizer or downcomer feedwater control valve may occur because of a mechanical or a control system failure. The failure will result in increase of feedwater flow to the economizer section of one steam generator.

This event is assumed to occur 60 times during the 60-year plant design life.
w. Inadvertent isolation of one main FW heater

The inadvertent isolation of one main feedwater heater may result from an operator error or mechanical failure. This event results in a decrease in feedwater temperature to the steam generator.

This event is assumed to occur 60 times during the 60-year plant design life.

x. Startup and coastdown of a reactor coolant pump at hot standby (HSB)

The startup and coastdown of a reactor coolant pump at HSB occur during each plant heatup and cooldown operation.

This event is assumed to occur 2,000 times during the 60-year plant design life.

y. Startup and shutdown of the shutdown cooling system at hot shutdown (HSD)

The startup and shutdown of shutdown cooling system (SCS) at HSD condition occur during each plant cooldown and heatup operations.

This event is assumed to occur 250 times during the 60-year plant design life.

z. Spurious startup of a safety injection pump during shutdown condition

Spurious startup of a safety injection pump during shutdown condition is considered to occur due to operator error or control system failure.

This event is assumed to occur 60 times during the 60-year plant design life.

aa. Spurious actuation of the pressurizer heaters at HSB

The spurious actuation of the pressurizer heaters may occur because of a mechanical or a control system failure. The failure is assumed to actuate all pressurizer heaters while the plant is operating at hot standby condition.

This event is assumed to occur 60 times during the 60-year plant design life.

ab. Plant heatup and cooldown

Plant heatup is defined as operations that bring the RCS from a condition where the reactor is subcritical and the RCS is at nearly ambient temperature and
atmospheric pressure to a condition where the system temperature and pressure are at their normal operating zero-power values. The temperature changes of 37.8 °C (100 °F) per hour that bound the heatup rate are conducted by four reactor coolant pumps (RCPs). The heatup rate is controlled by the shutdown cooling system and the steam generators. The heatup rate for the pressurizer is 93.3 °C (200 °F) per hour.

Plant cooldown is a series of operations that bring the RCS from a power operation condition to a cold shutdown condition in preparation for refueling or other maintenance operations. The cooldown operations represented by ramp changes in temperature of 37.8 °C (100 °F) per hour that bound the cooldown rate are performed by the steam generator steam dump and shutdown cooling operation. The cooldown rate for the pressurizer is allowed up 93.3 °C (200 °F) per hour.

Each event is assumed to occur 250 times each during the 60-year plant design life.

3.9.1.1.2 Service Level B Conditions

Service Level B conditions consist of eight event conditions as shown in Table 3.9-1. Event conditions are categorized in a similar way as in the safety analysis. Each event condition may include specific events. The following are the event descriptions for specific events considered in Service Level B conditions.

a. Decrease in FW temperature

A decrease in main feedwater temperature results in an increase in heat removal by the secondary system. A postulated failure in the feedwater train is assumed to result in a decrease in feedwater enthalpy.

This event is assumed to occur 20 times during the 60-year plant design life.

b. Increase in FW flow rate

An increase in feedwater flow results in an increase in heat removal from the secondary system. A failure in the main feed train is assumed to cause an increase in feedwater flow up to a maximum flow rate of about 160 percent of rated flow.

This event is assumed to occur 20 times during the 60-year plant design life.
c. Increase in steam flow rate

An increase in main steam flow results in an increase in heat removal from the secondary system. A failure in the secondary system is assumed to cause an increase in steam flow up to about 11 percent of the full-power steaming rate.

This event is assumed to occur 20 times during the 60-year plant design life.

d. Inadvertent opening of a main steam safety valve

An inadvertent opening of a main steam safety valve results in an increase in heat removal from the secondary system. The event is postulated to result in the increase in main steam flow while operating at full power.

This event is assumed to occur 10 times during the 60-year plant design life.

e. Loss of external load

A loss of external load (event) occurs due to the separation of the turbine/generator from the electricity distribution grid. When house load operation is operable, the plant is controlled by reactor power cutback system (RPCS) and steam bypass control system without a reactor trip for loss of external load. When the house load operation is not operable, the turbine is tripped for loss of external load, the turbine stop valve is closed, and the steam flow from the steam generator to the turbine is blocked. This event shows the loss of external load for an inoperable case of house load operation.

This event is assumed to occur 20 times during the 60-year plant design life.

f. Loss of condenser vacuum

A loss of condenser vacuum event results in a decrease in heat removal from the secondary system. A loss of condenser vacuum can be caused by the failure of the circulating water system providing condenser cooling, failure of the main condenser evacuator system to remove non-condensable gases, or excessive in-leakage of air through a turbine gland. The turbine is assumed to trip immediately upon the loss of condenser vacuum.

This event is assumed to occur 20 times during the 60-year plant design life.
g. Loss of non-emergency AC power to the station auxiliaries

A loss of non-emergency ac power to the station auxiliaries event results in a decrease in heat removal by the secondary system. This event may result from a complete loss of the external grid or a loss of the onsite AC power distribution system. Emergency power is still supplied to the plant by emergency diesel generators. The loss of non-emergency AC power to the station results in the loss of the buses that power the reactor coolant pumps, thereby causing all the pumps to coast down.

This event is assumed to occur 20 times during the 60-year plant design life.

h. Main steam isolation valve closure

A main steam isolation valve (MSIV) closure event results in a decrease in heat removal by the secondary system. An MSIV closure event is initiated by the closure of all the MSIVs that is the result of a spurious closure signal (e.g., spurious main steam isolation signal). This results in termination of both main steam and main feedwater flow because the main steam isolation signal closes the main feedwater isolation valves.

This event is assumed to occur 20 times during the 60-year plant design life.

i. Loss of normal feedwater flow

A loss of main feedwater flow event results in a decrease in heat removal from the secondary system. The loss of main feedwater flow is caused by losing one or both main feedwater pumps or by spurious signals in the feedwater control system. The limiting loss of main feedwater event is defined as the total loss of main feedwater to both steam generators. The loss of main feedwater flow results in a decreasing steam generator level and an increasing secondary pressure. These secondary variations cause a corresponding increase in primary system temperatures and pressures.

This event is assumed to occur 20 times during the 60-year plant design life.

j. Loss of forced reactor coolant flow
A loss of forced reactor coolant flow event results in a decrease in the RCS flow. The loss of forced reactor coolant could be initiated by a failure in the RCP auxiliary system or loss of non-emergency AC power.

This event is assumed to occur 20 times during the 60-year plant design life.

k. Natural circulation cooldown (HSB to HSD)

If the AC power to all non-safety systems including the RCPs is lost during power operation, the plant is tripped and cooled down in a natural circulation mode. The amount of time needed to achieve plant shutdown conditions will be greater. Only safety-related systems are used to cool down the plant. A natural circulation cooldown operation is necessary to cool down the RCS from the HSB to HSD condition at which the operation of shutdown cooling system is allowed.

This event is assumed to occur 10 times during the 60-year plant design life.

l. Uncontrolled CEA withdrawal at low power

An uncontrolled control element assembly (CEA) withdrawal at low power or subcritical condition results in a reactivity or power distribution anomaly. The CEA withdrawal at low power or subcritical conditions adds positive reactivity to the reactor core by removing regulating CEAs. This causes an increase in core power and heat flux, resulting in increasing core temperatures and pressures.

This event is assumed to occur five times during the 60-year plant design life.

m. Uncontrolled CEA withdrawal at high power

An uncontrolled CEA withdrawal at high power event causes a reactivity or power distribution anomaly.

This event is assumed to occur five times during the 60-year plant design life.

n. Control rod misoperation, RPCS inadvertent operation, or operator error

A single full-length CEA drop event causes a reactivity or power distribution anomaly. The CEA drop is caused by failure in the CEA drive mechanism, causing an initial insertion of negative reactivity.
This event is assumed to occur 50 times during the 60-year plant design life.

o. Loss of component cooling water to the letdown heat exchanger

The loss of component cooling water to the letdown heat exchanger event results in increase in RCS inventory. The letdown isolation valve closes on high letdown flow temperature to prevent CVCS equipment from being exposed to high-temperature conditions.

This event is assumed to occur 10 times during the 60-year plant design life.

p. CVCS malfunction that increases RCS inventory

A CVCS malfunction event results in an increase in RCS inventory. The limiting scenario with respect to increases in RCS inventory is the pressurizer level control system malfunction. During this event, the letdown line is isolated and the charging control valve is fully opened.

This event is assumed to occur 10 times during the 60-year plant design life.

q. Inadvertent opening of the pilot-operated safety relief valve (POSRV closes as expected)

The inadvertent opening of the POSRV is an incident that results in a decrease in RCS inventory. Only one POSRV is assumed to open. The RCS pressure stops to decrease as the POSRV recloses.

This event is assumed to occur 10 times during the 60-year plant design life.

r. Failure of small lines carrying coolant outside containment (letdown line break)

The double-ended break of a letdown line outside containment event results in a decrease in RCS inventory. Because of the pipe size, the consequences of a double-ended letdown line break outside containment bound all possible instrument or sample line breaks.

This event is assumed to occur 20 times during the 60-year plant design life.

s. Reactor coolant pump seal failure
An RCP seal failure event is set as a DBE for the design of the RCP bleed-off line.

This event is assumed to occur 10 times during the 60-year plant design life.

t. Loss of seal injection with loss of cooling water

A loss of seal injection with a loss of cooling water event is set as a DBE for the design of the RCP bleed-off line.

This event is assumed to occur five times during the 60-year plant design life.

3.9.1.1.3 Service Level C Conditions

There are no events classified as a Service Level C condition. The events traditionally known as Service Level C conditions are classified as other Service Level conditions based on the recent nuclear power plant industry data (for example, a loss of offsite power with natural circulation event is classified as a Service Level B condition and one steam generator tube failure event as a Service Level D condition). The frequencies of events traditionally categorized as a Service Level C condition are conservatively modified to be classified as a Service Level B condition for design purpose.

3.9.1.1.4 Service Level D Conditions

Service Level D conditions consist of six event conditions as shown in Table 3.9-1. Event conditions are categorized similar to the Service Level B conditions. Hypothetical accidents are considered as Service Level D conditions. The following are the event descriptions for specific events considered in Service Level D conditions.

a. Steam system piping failure

A main steam line break (MSLB) results in an increase in heat removal by the secondary system. A rupture in the main steam line is postulated to cause an uncontrolled blowdown of the steam generators until the main steam isolation valves (MSIVs) close upon the receipt of a main steam isolation signal (MSIS). If the steam line break occurs downstream of the MSIV, the closure of the MSIVs will terminate the primary system cooldown. If the steam line break occurs upstream of the MSIVs, the ruptured steam generator continues to blow down after MSIS, causing a greater cooldown of the primary system.
This event is assumed to occur one time during the 60-year plant design life.

b. Feedwater system line break (FWLB)

The feedwater system pipe break or feedwater line break (FWLB) is an accident that results in a decrease in heat removal from the secondary system. A break in the feedwater system piping is postulated to occur and cause a dependent loss of the main feedwater pumps. If the break occurs upstream of the reverse flow check valves, the thermal-hydraulic response will be similar to that of a loss of normal feedwater flow event. If the break occurs between the last reverse flow check valve and the steam generator, both steam generators blow down until the main steam isolation valves close following receipt of a main steam isolation signal.

This event is assumed to occur one time during the 60-year plant design life.

c. Reactor coolant pump rotor seizure

A single RCP rotor seizure with the loss of offsite power is an accident that results in a decrease in reactor coolant system flow rate. The RCP rotor seizure is caused by the seizure of either the upper or the lower RCP thrust-journal bearings.

This event is assumed to occur one time during the 60-year plant design life.

d. Reactor coolant pump shaft break

An RCP shaft break with the loss of offsite power is an accident that results in a decrease in reactor coolant system flow rate. A single reactor coolant pump shaft break is postulated to cause a low reactor coolant system flow trip by reactor protection system.

This event is assumed to occur one time during the 60-year plant design life.

e. Rod ejection accident

A CEA ejection is an accident that causes a reactivity or power distribution anomaly. The CEA ejection is postulated to be caused by a circumferential rupture of the CEA drive mechanism housing or nozzle. The rupture is assumed to allow the instantaneous ejection of the rod with the largest reactivity bite. The
subsequent rapid reactivity excursion will cause variations in the NSSS process parameters.

This event is assumed to occur one time during the 60-year plant design life.

f. Inadvertent opening of pilot-operated safety relief valve (POSRV fails to close)

The inadvertent opening of POSRV is an accident that results in a decrease in RCS inventory. The RCS pressure will decrease continuously as the POSRV fails to close. Hence, a reactor trip occurs due to the low pressurizer pressure. This event is a small break loss of coolant accident.

This event is assumed to occur one time during the 60-year plant design life.

g. Steam generator tube rupture (SGTR)

The steam generator tube rupture (SGTR) is an accident that causes a decrease in RCS inventory. It is postulated that a double-ended rupture of a single U-tube occurs penetrating the barrier between the reactor coolant system and the main steam system.

This event is assumed to occur one time during the 60-year plant design life.

h. Loss-of-coolant accident (LOCA) resulting from postulated pipe breaks within the RCS pressure boundary

The LOCA is an accident that results in a decrease in RCS inventory.

This event is assumed to occur one time during the 60-year plant design life.

i. Total loss of feedwater flow (TLOFW)

The total loss of feedwater flow event is a beyond design basis event. This event is initiated by a loss of main and auxiliary feedwater flow and results in decrease in RCS heat removal.

This event is assumed to occur one time during the 60-year plant design life.
3.9.1.1.5 Testing Conditions

Test conditions consist of six event conditions as shown in Table 3.9-1. The following are event descriptions considered in test conditions.

a. RCS hydrostatic test

The hydrostatic test is performed to provide reasonable assurance of the integrity of the RCS pressure boundary, its components, and associated unisolable piping systems at 125 percent of design pressure.

This event is assumed to occur 15 times during the 60-year plant design life.

b. Secondary hydrostatic test

The secondary hydrostatic test is performed to provide reasonable assurance of the integrity of the secondary side of steam generator, including the unisolable portion of the main steam, main feedwater, blowdown, recirculation, and auxiliary feedwater lines, at 125 percent of design pressure.

This event is assumed to occur 15 times during the 60-year plant design life.

c. RCS leak test

Whenever the RCS has been opened, an RCS leak test is conducted at the normal operating pressure. The RCS loop pressure is raised following the limitations curve of the plant hydrostatic test by control of charging and letdown flow.

This event is assumed to occur 200 times during the 60-year plant design life.

d. Secondary leak test

Whenever the secondary system has been opened, a secondary side leak test is conducted at the design pressure.

This event is assumed to occur 200 times during the 60-year plant design life.

e. SIS/SCS preoperational and maintenance test
SIS/SCS preoperational and maintenance test is conducted to provide reasonable assurance of the operability of SIS/SCS components. The preoperational test is a pre-core testing performed to provide reasonable assurance that the components and/or systems perform their intended functions. The maintenance test provides reasonable assurance that the components perform their intended functions after maintenance.

This event is assumed to occur 360 times during the 60-year plant design life.

f. SIS/SCS check valve operability tests

SIS/SCS check valve operability test is conducted to provide reasonable assurance of the operability of SIS/SCS check valves. The test is performed by safety injection pumps (SIPs) in refueling mode or shutdown cooling pumps (SCPs) in cold shutdown mode.

This event is assumed to occur 120 times during the 60-year plant design life.

3.9.1.2 Computer Programs Used in Stress Analyses

3.9.1.2.1 Code Class Systems, Components, and Supports

The following paragraphs provide a summary of the applicable computer programs used in the stress and structural analyses for ASME Code Class systems, components, and supports in the APR1400 design. The summaries include individual descriptions and applicability data. The computer codes used in these analyses have been verified in conformance with design control methods, consistent with the quality assurance program described in Chapter 17.

3.9.1.2.1.1 ABAQUS

The ABAQUS program is a general-purpose nonlinear finite element program with structural and heat transfer capabilities. ABAQUS is used for stress analysis of regions of vessels, piping, or supports that may deform plastically under prescribed loadings. It is also used for elastic analyses of complex geometries where the graphics capability enables a well-defined solution. The thermal capabilities of ABAQUS are used for complex geometries where simplification of input and graphical output are preferred.
ABAQUS is commercially available and has had sufficient use to justify its applicability and validity. See Reference 5 for information on ABAQUS.

3.9.1.2.1.2 PICEP

The PICEP program calculates the flow through a crack in a pipe. PICEP uses the simplified engineering approach for elastic-plastic fracture analysis for finding the crack opening displacement and area. Fluid calculation options include single and two-phase flow as well as allowance for friction. PICEP, commercial software, was developed by the Electric Power Research Institute (EPRI). See Reference 22 in Subsection 3.6.5 for information on PICEP.

3.9.1.2.1.3 CLEVER

CLEVER determines SG and snubber stroke, and building interface boundaries for the SG snubber lever system. The program verifies the kinematics of the snubber lever linkage systems based on input motions of the SG lug and detailed snubber lever system geometry. See Reference 6 for information on CLEVER.

3.9.1.2.1.4 HeadPR (Head Penetration Reinforcement Program)

The HeadPR computer program calculates the available reinforcement and the reinforcement that is needed for penetrations in the hemispherical heads. The technique described in ASME Section is used.

The HeadPR computer program is used to perform the preliminary sizing and reinforcement calculations for hemispherical heads in the reactor vessel.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Reference 7 for more information.

3.9.1.2.1.5 CEFLASH-4B

The CEFLASH-4B computer program calculates transient conditions resulting from a flow line rupture in a water/steam flow system. The program is used to calculate steam generator internal loadings following a postulated main steam line and main feedwater line break.
The program was verified by comparison of program results and the result of the CEFLASH-4A computer program. See Reference 8 for more information.

3.9.1.2.1.6 ANSYS

The ANSYS computer program is a large-scale, general-purpose, finite element analysis program for linear and nonlinear structural and thermal analyses, and additional descriptive information on this code is provided in Subsection 3.9.1.2.2.1.

The program is used in numerous applications for all components in the areas of structural, fatigue, thermal, and eigenvalue analysis. Analysis capabilities include static and dynamic; elastic, plastic, creep and swelling; small and large deflections; steady-state and transient heat transfer; and fluid flow.

The program has been verified by comparison with known theoretical solutions, experimental results, and by other calculated solutions. See Reference 9 for more information.

3.9.1.2.1.7 AFP2D

The AFP2D computer program uses the thermal stresses of two-dimensional axisymmetric structures resulting from an ANSYS program run. The program combines thermal stresses calculated for transient load steps with stresses due to pressure and external mechanical loads, and calculates primary plus secondary stresses, peak stresses, and their ranges of stress intensities and fatigue usage factors.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Reference 10 for more information.

3.9.1.2.1.8 TSPOST

The TSPOST computer program uses the stresses in the tubesheet of the SG resulting from two-dimensional axisymmetric model using an ANSYS program run. The program evaluates the primary stress by various pressure conditions imposed on the SG primary and secondary sides, and calculates range of primary plus secondary stress intensity and cumulative usage factors, by combining the thermal and pressure stresses.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Reference 11 for more information.
3.9.1.2.1.9  **AFPOST**

The AFPOST computer program uses the thermal stresses of two- and three-dimensional structure resulting from ANSYS program run. The program combines thermal stress calculated for transient load steps with stresses due to pressure and external mechanical loads, and calculates primary plus secondary stresses, peak stresses and their ranges of stress intensities and fatigue usage factors.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Reference 12 for more information.

3.9.1.2.1.10  **ATHOS3**

The ATHOS3 is a three-dimensional, two-phase flow distribution computer program with both steady-state and transient capabilities. Homogeneous and axial flow algebraic slip models are available. A typical geometric model includes approximately 1,000 cells but more detailed models are possible. Secondary fluid results include the velocity vector components in the three coordinate directions, pressure, temperature/quality, and density in each cell. The corresponding heat flux, tube wall temperature, primary temperatures, and the circulation ratio are also output. Vector and scalar plotting is available. In transient calculations, water level and steam flow are calculated as a function of time.

The program was verified by comparing the results to measured data from small-scale experiments, model SGs, and full-scale SGs (see Reference 13 for more information).

3.9.1.2.1.11  **PTXIG**

This program applies the procedures of Appendix G of ASME Section XI and the supplemental procedures in Welding Research Council Bulletin 175 to evaluate pressure vessels against failure. The program calculates the allowable internal pressure as a function of crack size, $RT_{NDT}$, and thermal conditions that are input by the user. The program was verified by comparison of program results and hand calculations (see Reference 14 for more information).

3.9.1.2.1.12  **PIPESTRESS**

PIPESTRESS (Reference 15) is a piping analysis program that is applied to the static and dynamic analyses including response spectra and time-history analyses.
PIESTRESS is used for the analysis of ASME Section III, Class 1, 2, and 3 (Reference 16) as well as ASME B31.1 and B31.3 piping systems (References 17 and 18).

3.9.1.2.1.13 REFORC – DEC

REFORC determines flow-induced forces in piping system by serving as a post-processor to a thermal-hydraulic transient code, RELAP5/MOD3.1. See Reference 19 for more information.

3.9.1.2.1.14 RELAP5/MOD3.1

RELAP5/MOD3.1 (Reference 20) is a best-estimate system code suitable for the analysis of all transients and postulated accidents in light water reactor (LWR) systems, including both large- and small-break loss-of-coolant accidents (LOCAs) as well as the full range of operational transients. The one-dimensional RELAP5/MOD3.1 code is based on a non-homogeneous and non-equilibrium model for the two-phase system. This code is also used to analyze rapid transients such as pipe breaks and valve quick opening, and to extract some thermal-hydraulic parameters of the results such as velocity, density, and pressure for generation of hydrodynamic loads on piping system using a force generation code, REFORC-DEC.

3.9.1.2.1.15 RELAP5/MOD3.3

RELAP5/MOD 3.3 is a best-estimate system code suitable for the analysis of all transients and postulated accidents in LWR system, including both large- and small-break LOCAs as well as the full range of operational transients. The one-dimensional RELAP5/MOD3.3 code is based on a non-homogeneous and non-equilibrium model for the two-phase system. This code is also used to analyze rapid transients such as pipe breaks and valve quick opening and some thermal-hydraulic parameters of the results such as velocity, density, and pressure are extracted for generation of hydrodynamic loads on piping system (see Reference 21 for more information).

3.9.1.2.1.16 NOZPROG

The NOZPROG computer program calculates the maximum stress intensities developed in nozzles and nozzle-vessel intersection due to combinations of various external pipe loads under pressure.
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The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Reference 22 for more information.

3.9.1.2.2 Reactor Internals, Fuel, and CEDMs

The following computer programs are used in the static and dynamic analyses of reactor internals, fuel, and CEDMs.

3.9.1.2.2.1 ANSYS

ANSYS is a general-purpose linear and nonlinear finite element program with structural and heat transfer capabilities, and is described in Reference 23. Finite element analyses of reactor internal structures such as flanges and the lower support structure are performed with ANSYS to determine vertical and lateral stiffnesses. The program is also used to perform the static and dynamic analyses of the reactor internals to determine its structural stress responses.

The developer, ANSYS, has published an ANSYS verification manual with numerous examples of its usage.

3.9.1.2.2.2 ASHSD

ASHSD is used to obtain the dynamic response of the core support barrel under normal operating conditions and a LOCA. The program yields the dynamic shell and beam mode response of the structural system.

ASHSD has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results. The details of these comparisons are provided in References 24 and 25.

3.9.1.2.2.3 CESHOCK

The CESHOCK program is used to obtain the transient response of the reactor internals and fuel assemblies due to pipe break and seismic loads.

The computer program CESHOCK determines the response of structures that can be represented by lumped-mass and spring systems subjected to a variety of arbitrary-type loadings. Further description is provided in Reference 26.
CESHOCK has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted analytical results via an independent computer code. The details of these comparisons are available in References 25 and 26.

### 3.9.1.2.2.4 CEFLASH-4B

The CEFLASH-4B computer code (Reference 27) predicts the reactor coolant system pressure and flow distribution during the subcooled and saturated portion of the blowdown period of a LOCA. The equations for conservation of mass, energy, and momentum, along with a representation of the equation of state, are solved simultaneously in a node and flow path network representation of the primary RCS.

CEFLASH-4B provides transient pressures, flow rates, and densities throughout the primary system following a postulated pipe break in the RCS.

The CEFLASH-4B computer code is a modified version of the CEFLASH-4A code (References 28 and 29). The CEFLASH-4A computer code has been approved by the NRC (References 30 through 32). The capability of CEFLASH-4B to predict experimental blowdown data is presented in Reference 27.

### 3.9.1.2.2.5 DPVIB

The DPVIB program is used to calculate fluctuating pressure distribution in the downcomer region caused by RCP pressure pulsation.

Main inputs of DPVIB are system pressure, temperature, and pump forcing frequency. The pressure distribution in the downcomer region is obtained by solving the acoustic wave equation using an analytical method described in Reference 39. The DPVIB program also generates Fourier coefficients that are used in the stress analysis of the CSB.

### 3.9.1.3 Experimental Stress Analyses

Experimental stress analysis is not used for the APR1400.
3.9.1.4 Consideration for the Evaluation of the Faulted Condition

3.9.1.4.1 Seismic Category I RCS Items

The major components of the RCS are designed to withstand the forces associated with the pipe breaks described in Section 3.6 in combination with the forces associated with the safe shutdown earthquake (SSE), in-containment refueling water storage tank (IRWST) discharge, and normal operating conditions. For structural evaluation, the pipe breaks are those breaks for which a leak-before-break (LBB) cannot be demonstrated. Since the dynamic effects of breaks in piping systems listed in Subsection 3.6.3 are eliminated by LBB, the pipe break loads analysis procedure considers only those branch line pipe breaks not eliminated by an LBB.

The resultant component and support reactions are specified for design verification by the methods described below and in Subsection 3.9.3.

The system or subsystem analysis used to establish or confirm loads that are specified for the design of components and supports is performed on an elastic basis. The details of the models and forcing functions are described in Appendix 3.9B.

When an elastic system analysis is used to establish the loads that act on components and supports, elastic stress analysis methods are also used in the design calculations to evaluate the effects of the loads on the components and supports. In particular, inelastic methods such as plastic instability and limit analysis methods, as defined in the ASME Section III, are not used in conjunction with an elastic system analysis. The RCS and associated supports, which are analyzed using elastic methods, are shown in Figure 3.9-1.

Inelastic methods of analysis are used in cases where deemed desirable and appropriate to permit significant local inelastic response. In these cases, if any, the system or subsystem analyses performed to establish the loads that act on components and component supports are modified to include the inelastic strain compatibility in the local regions of the components and component supports at which significant local inelastic response is permitted.

Inelastic methods defined in the ASME Section III as plastic instability or limit analysis methods are not used.
3.9.1.4.1.1 Non-Code Items

The components not covered by the ASME Code but related to plant safety include:

a. Reactor internal structures (Class IS)

b. Fuel

c. Control element drive mechanisms (CEDMs)

d. Control element assemblies (CEAs)

Each component is designed in accordance with the relevant criteria to provide reasonable assurance of its operability as related to safety. The fuel assembly and CEA design is described in Section 4.2. The non-code components of the CEDMs are proven by testing as described in Subsection 3.9.4.4.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Piping vibration, thermal expansion, and dynamic effects are tested during the initial test program (ITP) as delineated in Section 14.2. The ITP is implemented to verify that the piping for all ASME Section III Class 1, 2, and 3 piping systems, other high-energy piping systems inside seismic Category I structures, high energy portions of systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level, and seismic Category I portions of moderate-energy piping systems located outside containment will remain within acceptable limits when subjected to piping vibrations and dynamic transients such as those caused by in-line component trips.

The ITP is conducted in accordance with the ASME OM-S/G-1990 (Reference 70).

The ITP is implemented to demonstrate that these piping systems, restraints, components, and supports have been designed to withstand flow-induced dynamic loading under the steady-state and operational transient conditions anticipated during service, to confirm that proper allowance for thermal contraction and expansion is provided, and to demonstrate that piping vibrations are within the acceptable level. The supports and restraints
necessary for operation during the life of the plant are considered to be parts of the piping system.

The test procedure includes a list of systems, flow modes, and selected locations for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessive vibration or indications of thermal motion restraint occur.

The proper installation and operation of snubbers is verified through visual inspections, hot and cold position measurements, and observation of thermal movements during the ITP. A check of snubber operability is made by recording hot and cold positions and comparing these positions to calculated hot and cold positions. The list of snubbers on systems, which experience sufficient thermal movement to measure snubber travel from cold to hot position, will be provided as part of the test procedure. In addition, the test procedure includes the procedure necessary to verify the snubber operability when snubber travel is not measured.

During the startup testing, the piping systems which are specified as part of the ITP are operated to check the performance characteristics of critical pumps, valve, controls, and auxiliary equipment. The flow modes of operation and transients that are performed during the test include pump trips and valve closures. In the case of the RCS heatup tests, transients that are applied to the system include the RCP start, RCP trip, the operation of pressure-relieving valves, and closure of a turbine stop valve. Additional requirements of startup testing are outlined in Subsection 14.2.1.

The general requirements for vibration and dynamic effects testing of piping systems are specified in NRC RG 1.68 (Reference 69). Test specifications are in accordance with ASME OM-S/G-1990, Part 3 (Reference 70). If vibration is noted beyond the acceptance levels, corrective measures to the restraints are designed, incorporated in the piping systems analysis, and installed. If the test results determines that the piping system restraints are inadequate or are damaged, corrective measures to the restraints are installed and another test is performed to determine whether the vibrations have been reduced to an acceptable level.

3.9.2.1.1 Steady-State Vibration

The above piping systems in Subsection 3.9.2.1 with the potential to exhibit significant vibration are monitored for steady-state vibration.
The details relating to this test are described in the test procedure prepared in accordance with ASME OM-S/G-1990, Part 3. The piping is monitored for normal operating and test modes along with operating modes expected to result in the most severe vibration. The piping is visually inspected and vibration movements are measured using portable instrumentation at locations where the vibration is determined to be the most severe. The piping, if necessary, is instrumented and monitored remotely.

The measured piping displacements are compared with allowable displacement limits that are based on the allowable stress amplitudes, $S_{alt}$, calculated in accordance with ASME OM-S/G-1990, Part 3. $S_{alt}$ is limited as defined below.

\[ S_{alt} = \frac{C_2 K_2 M}{Z} \leq \frac{S_{el}}{\alpha} \]

Where:

- $C_2$ = secondary stress index as defined in ASME Section III
- $K_2$ = local stress index as defined in ASME Section III
- $M$ = maximum zero to peak dynamic moment loading due to vibration only, or in combination with other loads, as required by the system design specification
- $S_{el}$ = 0.8 $S_A$, where $S_A$ is the alternating stress at $10^6$ cycles in psi from ASME Section III, Appendices, Figure I-9.1; or $S_A$ at $10^{11}$ cycles from the ASME Section III, Appendices, Figure I-9.2.2. The user considers the influence of temperature on the modulus of elasticity.
- $Z$ = section modulus of the pipe
- $\alpha$ = allowable stress reduction factor: 1.3 for materials covered by the ASME Section III, Appendices, Figure I-9.1; or 1.0 for materials covered by the ASME Section III, Appendices, Figure I-9.2.1 or I 9.2.2
For ASME Section III, Class 2 and 3 piping systems and ASME B31 piping systems

\[ S_{\text{alt}} = \frac{C_2K_2}{Z} M \leq \frac{S_{\text{el}}}{\alpha} \]

Where:

\[ C_2K_2 = 2i \]

\[ i = \text{stress intensification factor, as defined in ASME Section III, NC and ND or ASME B31} \]

The allowable stress reduction factor provides reasonable assurance that the alternating stress \( S_{\text{alt}} \) is based on the number of cycles during the design life.

If the measured piping displacements exceed allowable limits, one or more of the following actions are taken so that the vibration can be qualified.

a. Analyses are performed to show that the measured displacements are acceptable.

b. Additional testing is performed to show that the peak stresses due to the vibration are acceptable.

c. The source of the excessive vibrations is eliminated.

d. The pipe supporting arrangement is modified to reduce the vibration within acceptable levels.

3.9.2.1.2 Dynamic Transient Vibration

The dynamic transient vibration differs from steady-state vibration in that it occurs in a relatively short period of time and is usually generated by much larger forces.

In piping systems, the dynamic transient vibrations are most evaluated on the basis of pipe deflections and reactions. The primary cause of the dynamic transient vibrations is a high-pressure or low-pressure pulse traveling through the fluid. The dynamic transient vibrations are usually induced by rapid start or trip of a pump or turbine, and the quick closing or opening of valves such as turbine-stop valves and various types of control valves. The dynamic transients also occur as a result of rapid actuation of safety/relief valve (SRV)
opening or as a result of unexpected events, such as condensed water accumulating at a low point in steam piping during a plant outage.

The operational dynamic transient condition having significant impact on the piping system is included in the test. The piping system is verified to operate appropriately by monitoring piping and pipe supports, including snubbers, subjected to the effects of the operational dynamic transient condition during the test.

The piping is instrumented to measure the system response during the dynamic transient events. The measured responses are compared with analytically predicted values from the piping stress reports.

If excessive system vibration is apparent during the dynamic transient events, an evaluation is performed to determine the cause and to identify the corrective action.

Alternatively, an analysis may be performed to demonstrate that the measured dynamic transient vibration does not cause the piping system in question to exceed stress or fatigue acceptance criteria.

3.9.2.1.3 Thermal Expansion

For piping systems expected to be subjected to significant thermal movements, the thermal expansion test is performed to verify that the piping system expands and contracts within the acceptable limits based on analytically predicted movements from the piping stress analyses during the ITP and is performed in accordance with the requirements of ASME OM-S/G-1990, Part 7. The detailed description of the thermal motion monitoring program will be included as part of the test procedure. The thermal motion monitoring program will include verification of snubber movement, adequate clearances and gaps, the acceptance criteria, and how the motion is to be measured.

Prior to heatup, the locations of potential thermal interferences are identified and appropriate corrective restraints are installed through a pre-heatup walkdown. One complete thermal cycle (i.e., cold to hot position and back to cold position) is monitored.

The piping and components are visually inspected and piping displacements are monitored at predetermined locations. The measurement locations are based on those of snubbers, hangers, and expected large displacements.
3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

3.9.2.2.1 Seismic Qualification Testing

The recommended guidance and requirements in NRC RG 1.100 (Reference 34) and IEEE Std. 344 (Reference 35) are used for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment. The seismic qualification testing methods for safety-related mechanical equipment are described in Subsection 3.10.2.2.

3.9.2.2.2 Seismic System Analysis Methods

The seismic system analysis methods (e.g., response spectra analysis, time-history analysis, equivalent static load analysis) are described in Subsections 3.7.2 and 3.7.3. The method of analysis for piping and supports is described in Section 3.12. Seismic analysis methods for mechanical equipment and supports use the guidelines in IEEE Std. 344 and Subsection 3.10.2 and 3.10.3. The seismic analysis of mechanical equipment is performed by vendors to provide a seismic qualification report that demonstrates the structural integrity in accordance with the requirement of the design specification. The RCS seismic analysis is described in Appendix 3.9B.

3.9.2.2.3 Determination of Number of Earthquake Cycles

The OBE is chosen as 1/3 of the SSE for the APR1400 (see Section 3.7). When the OBE is less than or equal to 1/3 SSE, design or analysis is not required for the OBE.

With the elimination of the OBE, to account for fatigue in analysis and testing, the guidance for determination of the number of earthquake cycles described in SECY-93-087 (Reference 36) is used to account for fatigue in analysis and testing. For piping analysis, the number of earthquake cycles to be considered is defined in Subsection 3.7.3.1.

Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Std. 344-1987.
3.9.2.2.4  **Basis for Selection of Frequencies**

The seismic analysis is accomplished to account for the resonant frequencies and the seismic responses of structures, subsystems, and components in their design. The fundamental frequency of restraints and supports system is designed to be greater than the ZPA frequency of the applicable ISRS, to ensure no additional amplification of the seismic loads on the restrained/supported equipment and components. The seismic responses of equipment and subsystems are maintained within the established limits. If the natural frequencies of the equipment and supporting structures are in the same range where resonance can occur, the resonance is considered for the seismic design.

3.9.2.2.5  **Three Components of Earthquake Motion**

The combination of three directional components of earthquake motion is in accordance with NRC RG 1.92 (Reference 37) as described in Subsections 3.7.2.6 and 3.12.3.2.

3.9.2.2.6  **Combination of Modal Responses**

The combination of modal responses is applicable when the response spectrum method of analysis is used, because the phase relationship between various modes is not identified and only the maximum responses for each mode are determined. Modal responses are combined by the methods described in Subsection 3.7.2.7 for the mechanical equipment.

For piping analysis, the guidance on combining the individual modal results in NRC RG 1.92 is used as described in Subsection 3.12.3.2.

3.9.2.2.7  **Analytical Procedures for Piping**

All seismic Category I and II piping is analyzed for seismic effects as described in Subsection 3.12.3.

3.9.2.2.8  **Multiple-Supported Equipment Components with Distinct Inputs**

When the equipment or component is supported at points with different elevations within a building and between buildings, either the envelope of these elevation response spectra or multiple supports excitation is used for the seismic qualification of the equipment.
If ISM (Independent Support Motion) method is utilized for alternate method of multiple supports excitation, the criteria for the use of ISM method will be followed in accordance with NUREG -1061, Volume 2, Section 4.

For analyzing the piping systems supported at multiple locations within a single structure or multiple structures, the method used is described in Subsection 3.12.3.2. The SSCs design procedure for differential settlement and relative displacement is described in Subsection 3.8.5.8.

3.9.2.2.9 Use of Constant Vertical Static Factors

A constant static factor is not used for the seismic design of seismic Category I structures, systems, and components specified in Subsections 3.7.2.10 and 3.7.3.6.

3.9.2.2.10 Torsional Effects of Eccentric Masses

All concentrated loads in a piping subsystem, such as valves and valve operators, are modeled as massless members with the mass of each component lumped at its center of gravity. Massless members are modeled by connecting the center of gravity of components to the centerline of piping so that the torsional effects of the eccentric masses are considered.

Torsional effects of eccentric masses are also considered in the analysis of seismic Category I subsystems other than piping.

3.9.2.2.11 Buried Seismic Category I Conduits, and Tunnels

The seismic criteria and methods used to analyze buried seismic Category I conduit, and tunnels are addressed in Subsections 3.7.3.7.

3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

Interaction of other piping with seismic Category I piping is addressed in Subsection 3.12.3.7.

3.9.2.2.13 Analysis Procedure for Damping

The damping values used for seismic analysis are consistent with NRC RG 1.61 (Reference 38) as described in Table 3.7-7.
3.9.2.2.14 Seismic Analysis for Mechanical Tanks

The structural integrity of tanks such as fire water tank, fuel tanks for the emergency diesel generator, and other mechanical tanks that are classified as seismic Category I is demonstrated by analysis. Structural analysis without testing is used if the structural integrity of such tanks can verify the intended design function.

The seismic analyses of the mechanical tanks are performed using a separate (decoupled) finite element model to determine their natural frequencies and mode shape. Seismic loads are calculated, depending on the natural frequency results from the finite element analysis, either by using the equivalent static method or dynamic method if the tanks are considered flexible (i.e., frequency less than 50 Hz). The impulsive and convective forces are also considered in the tank analyses. The stress evaluation of the mechanical tanks, bolts, tank plates, and tank supports are performed to ensure that the calculated stresses at all investigated locations are less than their corresponding allowable values.

The COL applicant is to review the detailed analysis of mechanical tanks, including the effects of fluid sloshing (COL 3.9(7)).

3.9.2.2.15 Test and Analysis Results

The test and analysis results are documented and available for review. The implementation program that includes milestones and completion dates is further described in Section 3.10.

3.9.2.3 Dynamic Response Analysis for Reactor Internals under Operational Flow Transients and Steady-State Conditions

The flow-induced vibration of the reactor internals components during normal operation can be characterized as a forced response to both deterministic and random pressure fluctuations in the coolant. Methods have been developed to predict the various components of the hydraulic forcing function and the response of the reactor internals to such excitation.

This analytical methodology is summarized in Figure 3.9-2. The method separates the response calculations into two groups in accordance with the physical nature of the loading. Methods for developing the deterministic component of the hydraulic forcing function are
described in Subsection 3.9.2.3.1.1, while those relating to the random component are described in Subsection 3.9.2.3.1.2.

The responses of the reactor vessel core support and internal structures, including core support barrel assembly, upper guide structure assembly, and lower support structure assembly, to the normal operating hydraulic loads are calculated by finite element techniques. The mathematical models used in these response analyses are described in Subsection 3.9.2.3.2. The methods used in calculating the structural responses are described in Subsection 3.9.2.3.3.

3.9.2.3.1 Hydraulic Forcing Function

3.9.2.3.1.1 Deterministic Forcing Function

An analysis based on a hydrodynamic model is used to obtain the relationship between RCP pulsations in the inlet ducts and the deterministic pressure fluctuations on the core support barrel. A detailed description of this model and subsequent solution are given in References 39 and 40. The model represents the annulus of coolant between the core support barrel and the reactor vessel. In deriving the governing hydrodynamic differential equation for the model, the fluid is taken to be compressible and inviscid. Linearized versions of the equations of motion and continuity are used. The excitation on the hydraulic model is harmonic with the frequencies of excitation corresponding to pump rotational speeds and blade passing frequencies.

The dynamic force on the upper guide structure assembly is due to flow-induced forces on the tube bank. The deterministic components of these forces are caused by pressure pulsations at harmonics of the pump rotor and blade passing frequencies, and vortex shedding due to crossflow over the tubes.

The in-core instrumentation (ICI) nozzles and the skewed beam supports for the ICI support plate of the lower support structure are excited by deterministic and/or random, flow-induced forces. The deterministic component of this loading is due to pump-related pressure fluctuations and vortex shedding due to crossflow.

Data from the System 80 preoperational test (References 41 and 42) is used to determine the magnitude of these pulsations at the pump rotor and blade passing frequencies and their harmonics.
3.9.2.3.1.2 Random Forcing Function

The random hydraulic forcing function is developed based on the System 80 preoperational test (Reference 42). The forcing function is represented in the form of power spectral density together with associated coherence area. The forcing function is modified to reflect the flow rate and density differences based on an analytical expression found in Reference 43.

At normal operating conditions, the shroud tubes of upper guide structure assembly are excited by an upstream wake produced by turbulent buffeting. The forcing function for this type of loading can be represented as a band-limited white noise power spectrum.

The ICI nozzles and ICI support plate support beams of the lower support structure assembly are both subject to turbulent buffeting by the flow skirt jets. The outermost ICI nozzles and beams receive full impact of the jets before the jets decay due to fluid entrainment and the presence of inner tube rows. The force spectrum of these jets is assumed to be represented as wideband white noise. The magnitude of this spectrum is computed based on data from the System 80 preoperational tests.

3.9.2.3.2 Mathematical Models

Finite element analyses are performed on each of the reactor internals components using mathematical models. These models are designed to provide the most efficient analysis under the most significant loading condition to which each structure is exposed. The core support barrel (CSB) assembly is modeled as a shell using the ASHSD computer code as shown in Figure 3.9-3. The structure is fixed at the upper flange to determine the beam modes and frequencies. The shell modes and frequencies are found by considering the upper flange fixed and the lower flange pinned. These analyses include hydrodynamic mass effects. All significant mode shapes and frequencies are used in combination to perform the normal operating deterministic response analysis. A simplified finite element model of the barrel assembly is generated in the ANSYS program for use in the random response analysis.

The inner barrel assembly is modeled as plate and solid elements using the ANSYS computer code as shown in Figure 3.9-4. The model is used to determine the static and dynamic characteristics as well as periodic and random response analyses.
The control element assembly (CEA) guide tubes, fuel alignment plate and UGS support plate are modeled as beam and plate elements using the ANSYS code (Figure 3.9-5). The model includes details in the regions of selected CEA guide tubes and HJTC tubes to analyze detailed responses of these tubes.

The lower support structure is modeled as plate elements using the ANSYS computer code to determine the modes, natural frequencies, and responses. The in-core instrumentation assembly (Figure 3.9-6) is modeled as plate, beam, mass, and spring elements using ANSYS to analyze the dynamic characteristics and responses.

3.9.2.3.3 Response Analysis

3.9.2.3.3.1 Periodic Response

The normal mode method (Reference 44) is used to obtain the structural response of the reactor internals to the periodic forcing functions developed in Subsection 3.9.2.3.1.1. The method is applied to the appropriate finite element models described in Subsection 3.9.2.3.2. Generalized masses based on mode shapes and the mass matrices are calculated for the modes of vibration of each component. Modal force participation factors are based on the mode shapes and the predicted periodic forcing functions are calculated for each mode and forcing function. The generalized coordinate response for each mode is then obtained from the solution of the corresponding set of independent second order single-degree of freedom equation. Using mode shapes, the modal responses of the reactor internals are obtained by means of the appropriate coordinate transformations. Response to any specific forcing function is obtained through summation of the component modes for that forcing function.

3.9.2.3.3.2 Random Response

The normal mode method is used to obtain the structural response of the reactor internals subjected to random forcing functions. The random forcing functions are assumed to be of both the band limited and wide band white noise varieties as described in Subsection 3.9.2.3.1.2. Experimental and analytical expressions are used to define the force power spectral density associated with flow related turbulence and jet impact. The appropriate mathematical models described in Subsection 3.9.2.3.2 are used in the ANSYS. This code computes the root mean square (RMS) displacements, loads, and stresses in a multi-degree-of-freedom linear-elastic structural model subjected to stationary random dynamic loadings, such as those described in Subsection 3.9.2.3.1.2.
A value of 3 × RMS is used for considering peak responses to random loading. These peak values are then combined with results from other analyses and used in design verification analyses. The use of the value 3 × RMS is common design practice based upon the assumptions of Random Gaussian loading of structures made of ductile materials as described in Reference 45.

3.9.2.4 Preoperational Flow Induced Vibration Testing of Reactor Internals

In accordance with NRC RG 1.20 (Reference 46), a comprehensive vibration assessment program (CVAP) is conducted for reactor internals. The detailed evaluation of the CVAP for the APR1400 reactor internals, including the SG internals, the RCS piping and the piping attached to the RCS, is provided in Technical Report, APR1400-Z-M-NR-14009-P (Reference 47). The APR1400 is classified as non-prototype Category I, per NRC RG 1.20, with Palo Verde Unit 1, a Westinghouse System 80 reactor, as the valid prototype (Reference 42). According to CVAP Report of System 80 reactor internals (Reference 42), evaluation of the comparisons of analytical predictions, test measurements, and visual inspection results leads to the conclusion that the System 80 prototype reactor internals are structurally adequate and acceptable for long-term operation. The Palo Verde Unit 1 design and the APR1400 design are substantially the same with regard to arrangement, design, size, and operating conditions. The comparisons of arrangement, design, size, and operating conditions for reactor internals are provided in Tables 3.9-15 through 3.9-17.

The CVAP for the APR1400 design consists of the analysis program and inspection program. The analysis program is described in Subsection 3.9.2.3.

The inspection program consists of a pre-hot functional test inspection and a post-hot functional test inspection of the reactor internals. The duration of the hot functional test is established to provide reasonable assurance that $10^6$ cycles of vibration have occurred before the post-hot functional inspection. A detailed inspection of major load-bearing surfaces, contact surfaces, welds, and maximum stress locations identified in the analysis program is performed. All observations made during the pre-hot and post-hot functional test inspections are documented. A comparison of the structures is made to verify that no loss in structural integrity due to flow-induced vibrations has occurred.

The evaluation, consisting of the analysis program and inspection program, confirms the adequacy of the analysis prediction techniques and the structural integrity of the APR1400 design according to the guidance of NRC RG 1.20. The COL applicant is to provide the
inspection results for the APR1400 reactor internals classified as non-prototype Category I in accordance with NRC RG 1.20 (COL 3.9(1)).

NRC RG 1.20 recommends that the potential adverse effects from pressure fluctuations and vibrations in piping systems be considered for the steam generator internals in PWRs, which may be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. The steam dryers and separators in the APR1400 SG are subject to secondary side steam flow. Although there are instances of similar components in boiling water reactors (BWRs) exhibiting excessive vibration resulting from plant power uprate, none have been reported for PWR SG designs to date. This is further supported by the Operational Performance Information System for Nuclear Power Plants (OPIS) database, which also does not have any incidents related to the flow-induced vibration problems for PWR SG upper internals.

The design of the APR1400 SG upper internals and the flow conditions for which they are subjected are similar to the existing and currently operating SGs in the Republic of Korea and around the world. Based on the operational experience for the SG upper internals, KHNP concludes that these non-safety-related components will not exhibit excessive vibration. Consequently, no startup testing with measurement is planned for these components.

The vibration of the piping attached to the RCS as well as SG (main steam and main feedwater lines) is measured during initial startup testing. Strain gauges and accelerometers will be used to measure the vibration at selected locations for both steady-state and transient flow conditions. With regards to the condensate system, hand held vibration measurement equipment is used to measure accelerations at discrete locations in the piping system during startup testing.

The acceptance criteria for the piping will be based upon satisfying the appropriate displacement, acceleration, stress, and fatigue. The acquired data will be used to confirm that unexpected, abnormal vibrations do not occur, and that the vibration responses of the piping are sufficiently small compared to an acceptance criterion based on the design fatigue curves in ASME Section III.
3.9.2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

Dynamic analyses are performed to determine the maximum structural responses of the reactor internals, including the core to postulated pipe breaks and seismic loadings, and to verify the adequacy of their design.

The results of the analyses are required to meet the stress limits of ASME Section III, NG, for core support structures and the functional requirements of the reactor internals design specification. More information on the reactor internals design is provided in Subsection 3.9.5.3.

3.9.2.5.1 Seismic Analysis of Reactor Internals

The seismic analyses of the reactor internals including the core are performed separately for the horizontal and vertical directions.

In the horizontal direction, because the relative displacements between the core and core shroud and between the core support barrel and reactor vessel snubbers are sufficiently large to close the gaps that exist between these components, nonlinear horizontal time-history analyses are performed. The horizontal nonlinear analyses are divided into two parts. In the first part, the internals and core are analyzed to obtain the response of the internals and the proper dynamic input for the reactor core analysis. In the second part, the core plate motion from the first part is applied to a more detailed nonlinear model of the reactor core. The input excitation to the internals model is the response time-history of the reactor vessel at the internals support determined from the RCS analysis. Coupling effects between the internals and reactor vessel are accounted for by including a simplified representation of the internals with the RCS model.

A nonlinear analysis is also performed in the vertical direction. Possible lifting of the core off the core support surface, friction between the fuel rods and spacer grids, and fuel assembly holddown force are examples of nonlinear effects included in the vertical analysis.

In these analyses, two horizontal components and one vertical component of the seismic excitation are considered and the maximum responses for the three components are combined by the square root of the sum of the squares (SRSS) method.
The multi-mass mathematical models are developed to represent the reactor internals and core. The mathematical models of the internals are constructed in terms of lumped masses and linear elastic-spring elements. At appropriate locations within the internals and core, nodes are chosen to lump the weights of the structures. The criterion for choosing the number and location for mass concentration is to provide accurate representation of the significant modes of vibration of each of the internals components. The moments of inertia, cross-sectional areas, effective shear areas, and lengths are calculated for the element properties. The model definitions employ the procedures established in Westinghouse Topical Report CENPD-178 (Reference 48) and include hydrodynamic coupling effects. Separate horizontal and vertical models of the internals and core are formulated to more efficiently account for structural differences in these directions. In the horizontal nonlinear lumped mass representation of the internals and core, shown in Figure 3.9-15, gap and spring elements are used to represent contact between the fuel and core shroud. Lumped-mass nodes in the core are positioned to coincide with fuel-spacer grid locations. To simulate the nonlinear motion of the fuel, nonlinear spring couplings are used to connect corresponding nodes to the fuel assemblies and core shroud. The core is modeled by subdividing it into fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading.

The horizontal nonlinear reactor core model, consisting of one row of 17 individual fuel assemblies, is depicted in Figure 3.9-16. In this model, each fuel assembly is represented as mass points located at spacer grid locations. To simulate the gaps in the core, nonlinear spring couplings are used to connect corresponding nodes on adjacent fuel assemblies and the core shroud. The impact stiffness and impact damping parameters for the gap elements are derived from the impact tests, which are described in Section 4.2. The spacer grid impact representation used for the analysis is capable of representing two types of fuel assembly impact situations. In the first type, only one side of the spacer grid is loaded. This type of impact occurs when the peripheral fuel assembly hits the core shroud, or when two fuel assemblies strike one another. The second type of impact loading occurs typically when the fuel assemblies pile up on one side of the core. In this case, the spacer grids are subjected to a through-grid compressive loading.

The fuel assemblies in the coupled core/internals model and the detailed core model are modeled with beam elements to represent the horizontal stiffness between mass points and rotational springs at each end to simulate the end fixity existing at the top and bottom of the core. The value used for fuel horizontal stiffness and end fixity is based upon a parametric
study in which analytic predictions are correlated with fuel assembly static and dynamic test data. Fuel assembly structural damping as a function of vibrational amplitude is derived from fuel assembly forced vibration and pluck tests. The damping values used in the seismic analysis of the reactor internals are in accordance with the values in Table 3.7-7.

The vertical nonlinear model is shown in Figure 3.9-17. The vertical model stiffness values are generally calculated using bar characteristic equations. Nonlinear couplings are included between components to account for structural interactions such as those between the fuel and core support plate, and between the core support barrel and upper guide structure upper flanges. Preloads, which are caused by the combined action of applied external forces, deadweights, and holddown forces, are also included. Friction elements are used to simulate the coupling between the fuel rods and spacer grids.

It has been shown both analytically and experimentally (Reference 49) that immersion of a body in a dense-fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to those in air. The effect is more pronounced when the confining boundaries of the fluid are close to the vibrating body as in reactor internals. Hydrodynamic mass is employed to account for the effects of surrounding fluid on a vibrating system. The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass and the space between the real mass and confining boundary. Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in References 49 and 50.

The nonlinear seismic response and impact forces for the internals and fuel are determined using the CESHOCK computer program (see Subsection 3.9.1.2.3). The computer programs provide numerical solutions to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model is the time-history acceleration of the reactor vessel.

Input to the computer program consists of initial conditions, nodal lumped masses, linear-spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time histories. The output from the computer program consists of displacements, velocities, accelerations, impact forces, axial forces, shears, and moments.

The procedures used to account for damping in the analysis of the reactor internals and core are provided in Subsection 3.7.2.15.
The nonlinear response loads for the internals, including impacting, are determined for the vertical and horizontal directions. Vertical loads for the fuel are determined in the coupled model of internals and core and horizontal loads for the fuel are determined in a separate reactor core nonlinear analysis. The results are determined for the SSE.

3.9.2.5.2 Pipe Rupture Analysis of the Reactor Internals

3.9.2.5.2.1 Input Excitation

According to the application of LBB, the postulated pipe breaks for the reactor coolant system, reactor internals, and fuel assembly are defined in Subsection 3.6.2. The secondary side breaks, which cause vibratory motion to the reactor internals, do not cause blowdown loads within the reactor vessel. However, the primary side break causes blowdown loads as well as vibratory motion. The blowdown loads consist of transient pressure, flow rate, and density distributions throughout the primary RCS.

The transient pressure, flow rate, and density distributions are computed for the subcooled and saturated portions of the blowdown period during the pipe breaks. The computer code that is used is based on a node flow path concept in which control volumes (nodes) are connected in any desired manner by flow areas (flow paths). A complex node / flow path network is used to model the reactor coolant system. The modeling procedure has been compared to a large-scale experimental blowdown test with excellent agreement. The laws of conservation of mass, energy, and momentum along with a representation of the equation of state are solved simultaneously. The hydraulic transient of the reactor is coupled to the thermal response of the core by analytically solving the one-dimensional radial heat conduction equation in each core node. Pre-blowdown steady-state conditions in the reactor coolant system are established through the use of specified input quantities. The blowdown loads model uses a non-equilibrium critical flow correlation for computing the subcooled and saturated critical fluid discharge through the break.

A break in the reactor coolant system results in large local pressure differences across various reactor internal components and an acceleration of the local fluid velocity in various regions. The acceleration of the local fluid velocity can result in higher component drag loads than during steady-state reactor operation.

The total instantaneous load across the core is given by the summation of the pressure forces acting in the direction of the pressure gradient and the drag forces acting parallel to the flow. The loads are obtained using a control volume approach using an integrated
fluid momentum equation. The drag forces are represented by the fluid shear term in this equation and consist of both frictional and form drag.

3.9.2.5.2.1.1 Reactor Internals Asymmetric Blowdown Loads

During an inlet break, dynamic lateral loads are developed on the core support barrel by the time varying radial pressure disturbances. These are highly asymmetric in the circumferential direction and are caused by the expansion of wave fronts and flow redistributions acting on the surface of the barrel. In order to obtain these applied lateral loads, the time-dependent Fourier coefficients that define the circumferential pressure distributions are first obtained. A linear variation in pressure is assumed to act between each elevation. The sine and cosine pressure distributions produce a response in the barrel, which is analogous to a beam-bending mode. When integrated over the surface area of the core support barrel, these pressures produce resultant lateral loads that are applied to the model.

3.9.2.5.2.1.2 Control Element Guide Tube Loads

During normal operation, the reactor coolant flows axially through the core into the upper guide structure. Within the upper guide structure, the coolant flow changes direction so that it exits through the hot leg nozzles.

During an outlet break, the transverse flow of the coolant across the control element guide tubes gives rise to loads that induce deflections in these tubes. The transverse drag forces are determined from flow model experiments that are geometrically and dynamically similar to the full-scale upper guide structure design for the System 80.

3.9.2.5.2.1.3 Vertical Hydraulic Loads

The vertical hydraulic loads on the reactor internals due to postulated pipe breaks are considered. The loads are obtained using a control volume approach using an integrated fluid momentum equation. This is based on control volumes that are consistent with both fluid volumes and the lumped mass nodes of the vertical analysis model.

The resulting reactor internals blowdown loads are used in subsequent analyses to determine the structural response of the internals and fuel.
3.9.2.5.2.2 Analysis Model

The horizontal model of the reactor internals used for the pipe break analysis is the same as the seismic model (Figure 3.9-15) except for the control element guide tubes (tube bank) region. The outlet pipe breaks (see Subsection 3.9.2.5.2.1.2) result in direct application of hydraulic loads to the tube bank region; therefore, more detail is required in the model for the outlet pipe breaks than for the inlet pipe breaks. The same model with seismic analysis (Figure 3.9-17) is used for pipe break analysis in the vertical direction.

3.9.2.5.2.3 Analysis Method

The nonlinear pipe break response and impact forces for the internals and fuel are determined using the CESHOCK computer program. The computer programs provide the numerical solution to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model is the time-history acceleration of the reactor vessel.

Input to the computer program consists of initial conditions, nodal lumped masses, linear-spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time-histories. The output from the computer program consists of displacements, velocities, accelerations, impact forces, axial forces, shears, and moments.

A cold leg branch line break causes a pressure transient on the core support barrel (CSB) that varies circumferentially as well as longitudinally. The ASHSD finite element computer code is used to analyze the shell response of the CSB to the pressure transient from a cold leg branch line break. The CSB is modeled as a series of shell elements joined at their nodal points, as shown in Figure 3.9-3. The length of the elements in each model is selected to be a fraction of the shell attenuation length.

A damped equation of motion is formulated for each degree of freedom of the system. Four degrees of freedom for radial displacement, circumferential displacement, vertical displacement, and meridional rotation are considered in the analysis. The differential equations of motion are solved numerically using a step-by-step integration procedure.

The circumferential variation of the pressure time-history is considered by representing the pressure as a Fourier expansion. The pressure at each node in the model is determined by linear interpolation. Thus, a complete spatial time load distribution compatible with the ASHSD computer program is obtained. Each load harmonic is considered separately by
ASHSD. The results for each harmonic are then added to obtain the nodal displacements, resultant shell forces, and shell stresses as a function of time.

3.9.2.5.3 In-Containment Refueling Water Storage Tank Discharge Analysis

The reactor internals and core are analyzed for the IRWST discharge loads.

The analytical models and the procedures of the reactor internals and core and the detailed core for IRWST discharge loads are the same as those used in the seismic analyses.

The input excitation to the reactor internals and core model is the acceleration time histories of reactor vessel flange and snubbers determined from the RCS IRWST analysis.

The maximum stresses resulting from IRWST, pipe break, and SSE are combined using the SRSS method to obtain the total stress intensities.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The results of the dynamic analysis of the reactor internals are compared with the results of preoperational tests of the prototype reactor (Reference 42) since the APR1400 reactor internals are classified as non-prototype Category I as described in Subsection 3.9.2.4.

The dynamic analysis models for the faulted condition are developed using the dynamic characteristics measured in the test of the prototype.

3.9.2.7 Dynamic System Analysis of the CEDM

The pressure-retaining components of the control element drive mechanism (CEDM) are designed to the appropriate stress criteria of ASME Section III for all loadings specified. The structural integrity of the CEDM for the seismic loadings is verified by combination of test and analysis. Methods of dynamic analysis using response spectrum analysis or time-history analysis are supported with experimentally obtained information.

3.9.2.7.1 Input Excitation Data

For the dynamic analyses, response spectra or time-history definition of the excitation at the base of the CEDM nozzle is obtained from the seismic analysis of the RCS. The
excitation is applied simultaneously in three mutually perpendicular directions (two horizontal and one vertical).

3.9.2.7.2 Analysis

A dynamic analysis of the mathematical structural model is performed using one or more of the computer programs described in Subsection 3.9.1.2.

3.9.2.7.3 Functional Test

Scram test using a minimum drop weight was performed by applying an incremental static deflection to the CEDM. From the test, the minimum radius of curvature of 2,025 inches for the upper pressure housing was obtained as the most critical criterion to ensure scramability. Deflection of the CEDM under seismic loading calculated by structural dynamic analysis was compared with the test result to verify scramability.

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Core Support Structures

This section describes the structural integrity of pressure-retaining components, component supports, and core support structures that are designed and constructed in accordance with the rules of the ASME Section III, Division 1 and GDC 1, 2, 4, 14, and 15.

The APR1400 design meets the SRP 3.9.3 criteria as described in the following respects:

a. 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1, as they relate to structures and components being designed, fabricated, erected, and tested to quality standards commensurate with the safety-related functions to be performed. These requirements provide reasonable assurance that safety-related components and structures meet service loading conditions, stress limits, and quality requirements of ASME Code permitted in 10 CFR 50.55a.

b. GDC 2 and 10 CFR Part 50, Appendix S, as they relate to structures and components important to safety being designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The effects of expected natural phenomena on the normal and accident conditions are considered in the loading combinations for structures and components important to safety.
c. GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents. The safety-related structures and components are protected against dynamic effects including LOCA by considering appropriate loading combinations as described in this section.

d. GDC 14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The RCPB components are designed in accordance with the ASME Code requirements that follow with GDC 14.

e. GDC 15 as it relates to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to provide reasonable assurance that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. These requirements provide reasonable assurance that the RCS and associated auxiliary system components meet ASME Code requirements.

f. 10 CFR Part 52 as it requires that components, component supports, and core support structures be designed and built in accordance with the certified design. Reasonable assurance that the design for the components, component supports, and core support structures is certified is provided by adhering to the applicable ASME Code requirements.

g. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, then a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC’s regulations. Reasonable assurance that the requirements for the components, component supports, and core support structures are met is provided by adhering to the applicable ASME Code requirements and ITAACs.

h. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning,
that the licensee will perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility is constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and NRC regulations. The proposed inspections, tests, and analyses are described in Section 14.3 and those applicable to emergency planning in Section 13.3.

In accordance with ASME Code, design specifications and design reports are to be provided for ASME Section III, Class 1, 2, and 3 components and piping. The design specification defines the jurisdictional boundary for the NF portion of the piping support. The supports are designed in accordance with ASME Section III, NF and meet its requirements.

Welding activities are performed in accordance with the requirements of ASME Section III. Component supports are fabricated in accordance with the requirements of ASME Section III, NF.

When dissimilar-metal welding joints are used for the fabrication of ASME Code Class components and piping, the weld materials and processes are selected in consideration of the joint degradation caused by the stress corrosion cracking in service environments.

Loading conditions, stress limits, design transients, and methods of analysis for ASME Code Class 1 reactor coolant loop piping and associated components and component supports are described in Subsection 3.9.3.1.

**3.9.3.1 Loading Combinations, Design Transients, and Stress Limits**

The loading combinations specified for the design ASME Section III, Class 1 components, supports, and piping are categorized as Design, Level A, Level B, Level C, and Level D conditions. The following specific loading combinations are specified for design:

**Design Loadings**

Mechanical loads are combined as defined in (a) and (b) below.

a. Design pressure plus deadweight plus IRWST discharge

b. Installation or handling loads (only for RV)
Level A Service Loadings

The following loading combination is considered as a Level A service loading.

Normal operation loads in combination with specified system operating transient loads resulting from the normal events.

In addition, handling loads alone at cold shutdown conditions are used to design only the reactor vessel.

Level B Service Loadings

Normal operation (including deadweight) and IRWST discharge loads in conjunction with the upset transients are considered Level B service loadings.

The seismic cycles mentioned in Subsection 3.7.3 are applied in a fatigue analysis, considering the effects of reactor coolant environment in accordance with NRC RG 1.207.

Level C Service Loadings

SRP 3.9.3 defines the design basis pipe break (DBPB) as those postulated pipe breaks other than a LOCA or MS/FPWB and the DBPB is identified as an emergency condition. This includes postulated pipe breaks in Class 1 branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.

Make-up flow can compensate for the loss of coolant from a break with a 5.56 mm (7/32 in.) internal diameter as described in Subsection 9.3.4. In accordance with the guidance in SRP 3.6.2, postulated breaks in one-inch nominal diameter piping and smaller piping do not require the analysis of the dynamic system loading from a ruptured pipe on components, component supports or core support structures. The DBPB condition also results in RCS temperature and pressure transient conditions and is thus conservatively included as a Level B condition in the RCS design transients given in Table 3.9-1. Therefore, Level C service loadings are not used in any APR1400 analyses.

Postulated breaks in lines larger than 25.4 mm (1 in.) nominal diameter are considered in pipe break analysis as described in Subsection 3.6.2 and included in the BLPB scope, which are treated as Level D conditions. The BLPB scope includes those postulated pipe breaks in the lines connected to the RCS that are not eliminated by LBB and that result in the loss of the reactor coolant at a rate in excess of the capability of the reactor coolant

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makeup system, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the RCS except those eliminated by LBB evaluation. The BLPB scope also includes main steam and main feedwater pipe breaks (MS/FWPB).

**Level D Service Loadings**

Normal operation (including deadweight), SSE, IRWST discharge, branch line pipe breaks, other accident loads, and faulted transients are combined as defined below.

a. Normal Operation + \([SSE^2 + (BLPB + IRWST)^2]^{1/2}\) in conjunction with the faulted transients

b. Normal Operation + \([SSE^2 + (POSRV + IRWST)^2]^{1/2}\) in conjunction with faulted transients (only for the pressurizer)

The SSE and pipe rupture loadings plus IRWST discharge loads are combined by the SRSS method in accordance with the guidelines of NUREG-0484, Rev. 1, 1980, or by a more conservative method.

**Test Loadings**

Test loadings are the concurrent test pressure in conjunction with hydrostatic deadweight.

The specific design transients specified for design are described in Subsection 3.9.1.1. The loading combinations and the stress limits associated with them, as defined in the ASME Code, also apply to the internals parts that are essential to the component in performing its safety function.

ASME Code Class 1, 2, and 3 piping and components of fluid systems are designed and constructed in accordance with ASME Section III. Hydrostatic testing is performed per ASME Section III.

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems are presented in the sections that describe the systems. The other loading conditions mean the loading conditions specified for the design of ASME Section III, Class 1, 2, and 3 components, supports, and piping, Control Element Drive Mechanism (CEDM), and Reactor Pressure Vessel Internals except for design pressure and temperature. These items are defined in Subsections 3.9.3.1, 3.9.4, and 3.9.5.
Stress analysis and fatigue evaluations are performed to determine structural adequacy of pressure components under the operating conditions of normal, upset, emergency, or faulted, as applicable.

Significant structural discontinuities in parts such as nozzles and flanges are considered. In addition to the design calculation required by ASME Section III, stress analysis is also performed by methods outlined in the code appendices.

Thermal stratification and striping are described not to create adverse effect on the associated piping in Subsection 3.12.5.10.

The loading combinations for the design of ASME Code Class 2 and 3 components and supports, CEDM, and reactor internals are described in Subsections 3.9.3.1.3, 3.9.4, and 3.9.5, respectively.

3.9.3.1.1 ASME Code Class 1 Components and Supports

Design transients for ASME Code Class 1 components and component supports are described in Subsection 3.9.1.1. Loading combinations for ASME Code Class 1 components are described in Table 3.9-2. Stress limits for ASME Code Class 1 components, supports, and piping are described in Table 3.9-3. The operating pressures of Code Class 1 active valves are limited to the pressures taken from the pressure-temperature rating of the ASME Section III for the maximum temperature of the given condition.

3.9.3.1.2 Reactor Internals

Design transients for reactor internals are described in Subsection 3.9.1.1. Loading combinations and stress limits are presented in Subsection 3.9.5.

3.9.3.1.3 ASME Code Class 2 and 3 Components and Supports

ASME Section III, NC and ND, provide design requirements for ASME Class 2 and 3 components. Loading combinations applicable to these Class 2 and 3 components and supports are described in Table 3.9-2. The stress criteria for the components are presented in Tables 3.9-5 through 3.9-9.

The Class 2 and 3 components that are subject to thermal or dynamic cyclic loads are evaluated for their fatigue sustainability using the ASME Section III, NC-3219.2 per SRP
3.9.3. A fatigue analysis is also performed in accordance with NC-3200 for the components that do not meet the NC-3219.2 criteria.

Stresses in valve bodies and pump casings are limited to the elastic limit of a material when the pump or valve is subjected to a combination of normal operating loads, seismic, and other applicable loads.

The functionality and operability of Class 2 and 3 safety-related active components are confirmed under combined service loadings at a stress level bounded by the specified service limit. The operability of active valves and pumps is described in Subsection 3.9.3.3.2.

In addition, the integrity of the pressure boundary for these components is properly maintained by verifying that calculated stresses are lower than their corresponding stress limits.

3.9.3.1.4 Preparation of Design Specifications

Design specifications for ASME Section III components and supports are prepared using guidelines given in Reference 34.

3.9.3.2 Design and Installation of Pressure Relief Devices

The pressure-relieving valves are designed in accordance with the requirements of ASME Section III, Appendix O. Where more than one valve is installed on the same pipe run, the sequence of valve openings to be assumed in analyzing for the stress at any piping location is estimated to induce the maximum instantaneous value of stress at that location. The applicable stress limits are satisfied for all components of the pipe run and connecting systems, and the pressure relief valve station, including supports. After the dynamic structural system analysis, a dynamic load factor affects the reaction forces and moments and a dynamic load factor of 2.0 is used in lieu of a dynamic analysis to determine the dynamic load factor.

3.9.3.2.1 Pressure Relief Devices Connected to the Pressurizer

The pressurizer pilot-operated safety relief valves (POSRVs) are designed to provide overpressure protection for the RCS. The POSRVs connected to the pressurizer are the only ASME Section III, Class 1 pressure relief valves in the APR1400.
Four pressurizer POSRVs are connected to the top of the pressurizer by separate inlet lines. There are two main discharge lines to the in-containment refueling water storage tank (IRWST). The steam from two POSRVs is discharged through one common discharge line. Each pressurizer POSRV provides the overpressure protection function with a main valve and two spring-loaded pilot valves in the assembly. The pressurizer POSRVs pass sufficient pressurizer steam to limit the RCS pressure to 110 percent of design pressure following a loss of load with delayed reactor trip, which is assumed to be initiated by the safety-grade signal from the reactor protection system (RPS). A delayed reactor trip is assumed to occur on a high-pressurizer pressure signal.

The pressurizer POSRVs and their pilot operators are qualified to operate in saturated steam, water, and steam and water mixtures in hot or cold conditions.

Details on the design of the POSRVs are provided in Subsections 5.2.2 and 5.4.14.

The opening of the POSRV introduces thermal-hydraulic transients in the discharge lines. The POSRVs discharge overpressurized reactor coolant into piping that carries the flow through a system of turns, straight segments, and area transitions or branches before expulsion at a remote location. Flow acceleration creates reaction forces.

For each main discharge line (two POSRVs), an analytical model is developed. The analytical model consists of the pressurizer, valves, and connecting piping from the pressurizer nozzles to the IRWST spargers. The pressurizer is modeled as a reservoir that is filled with saturated steam at constant pressure. For each straight pipe segment, a transient load is calculated. The analyses of the piping and support consider these reaction forces to be induced by the opening of the POSRVs.

3.9.3.2.2 Pressure Relief Devices for Class 2 Systems and Components

Pressure-relieving devices for ASME Section III, Class 2 systems include the main steam safety valves (MSSVs) and the main steam atmospheric dump valves (MSADVs) on the steam line and the low-temperature overpressure protection (LTOP) relief valves on the containment isolation portion of the normal shutdown cooling system (SCS).

The design and analysis requirements for the MSSVs, MSADVs, and discharge piping for the steam line are described in Subsection 10.3.2.
A relief valve on each of the SCS suction lines provides LTOP for RCS in a failure that initiates the pressure transient while in shutdown cooling and also can prevent overpressurization of the SCS. These relief valves are addressed in Subsection 5.4.7.

3.9.3.2.3 Pressure Relief Devices for Class 3 Systems and Components

Pressure-relieving devices for ASME Section III, Class 3 systems include relief valves and safety valves on heat exchangers, tanks, and piping lines to prevent overpressurization of the components and systems. The thrust load due to the valve opening is usually calculated using static analysis, and the load from static analysis is considered as input to the piping analysis with a dynamic load factor of 2.0. In case the static analysis produces undesirably conservative results, the valve thrust load is accomplished using dynamic analysis.

3.9.3.2.4 Pressure Relief Device Discharge System Design and Analysis

ASME Section III, Appendix O describes two types of discharge systems for pressure relief devices: open-discharge systems and closed-discharge systems. An open-discharge system discharges fluid directly to the atmosphere or to a vent pipe that is open to the atmosphere. A closed-discharge system is hard-piped to a distant location or closed tank. ASME Section III, Appendix O also describes the layout considerations and limits for both types of systems, as well as design equations and considerations for analysis of these systems. The APR1400 design conforms with these requirements.

3.9.3.3 Pump and Valve Functionality Assurance

3.9.3.3.1 Functionality Assurance Program

Active components are defined as pumps and valves and those components that perform a mechanical motion in order to shut down the plant, maintain the plant in a safe shutdown condition, or mitigate the consequences of a postulated event. Non-active components are those whose functional capability is not relied upon to perform a safety-related function for the various transients and plant conditions. The functional design and qualification of safety-related active components are performed in accordance with ASME QME-1-2007 (Reference 51) as accepted in NRC RG 1.100. The functional capability (performance of this mechanical motion) of active components during and after exposure to design basis events is confirmed by:
a. Designing each component to be capable of performing all safety functions during and following design basis events.

The design specification includes applicable loading combinations and conservative design limits for active components. The specification requires that the manufacturer demonstrate functional capability, by test, or by a combination of analysis and test.

b. Analysis and/or test demonstrating the functional capability of each design under the most severe postulated loadings.

Methods of functional capability demonstration programs are detailed in Subsections 3.9.3.3.2 and 3.9.3.3.3.

c. Inspection of each component to provide reasonable assurance of critical parameter conformance with specifications and drawings.

This inspection confirms that specified materials and processes are used, that wall thicknesses meet code requirements, and that fits and finishes meet the manufacturer’s requirements based on design clearance requirements.

d. Shop testing of each component to verify as-built conditions, as defined in Subsections 3.9.3.3.2 and 3.9.3.3.3.

e. Startup and periodic inservice testing in accordance with ASME OM Code as incorporated by reference in 10 CFR 50.55a to demonstrate that the active pumps and valves are in operating condition throughout the life of the plant.

The combined license (COL) applicant is to identify the site-specific active pumps (COL 3.9(2)).

3.9.3.3.2 Pump Functionality

ASME Class 2 and 3 safety-related active pumps are listed in Table 3.9-18. The following criteria are employed in a qualification program to ensure functional capability of the pumps required to function during and following design basis events.

a. Test or a combination of test and analysis are used in accordance with ASME QME-1-2007 as accepted in NRC RG 1.100 to confirm the adequacy of the pumps.
to function over the expected range of service conditions specified, including
design basis event and post-design-basis event conditions, as well as inservice
testing (IST) conditions.

b. The loads imposed by the attached piping along with the sustained dynamic and
seismic loads are taken into account. The design specification includes
applicable loading combinations and design stress limits for the pumps. In order
to provide reasonable assurance of functional capability under combined loadings,
the stresses resulting from the applied test loads envelop the specified service
stress limit for which the pump’s functional capability is intended. Design stress
limits applied in evaluating loading combinations are described in
Subsection3.9.3.1.

ASME Section III, Class 2 and 3 non-active pumps include the charging pumps, auxiliary
charging pump, and boric acid makeup pumps. Stress limits for non-active and active
pumps are shown in Table 3.9-6 and Table 3.9-7, respectively.

3.9.3.3.3 Valve Functionality

Safety-related active valves are listed in Table 3.9-4. ASME Class 1, 2, and 3 valves are
designed and analyzed according to the requirements of ASME Section III, NB/NC/ND-
3500.

The following criteria are employed in a qualification program to ensure functional
capability of the valves required to function during and following design basis events.

a. Test or a combination of test and analysis are used in accordance with ASME
QME-1-2007 as accepted in NRC RG 1.100 to confirm the adequacy of the valves
to function over the expected range of service conditions specified, including
design basis event and post-design-basis event conditions, as well as IST
conditions.

b. The loads imposed by the attached piping along with the sustained dynamic and
seismic loads are taken into account. The design specification includes
applicable loading combinations, and design stress limits for the valves. In order
to provide reasonable assurance of functional capability under combined loadings,
the stresses resulting from the applied test loads envelop the specified service
stress limit for which the valve’s functional capability is intended. Design stress
limits applied in evaluating loading combinations are described in Subsection 3.9.3.1.

The safety-related valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed:

a. Shell hydrostatic test to ASME Section III requirements

b. Backseat and main seat leakage tests

c. Disc hydrostatic test

d. Functional tests to verify that valve opens and closes as required when subjected to the design differential pressure and flow

Cold hydro qualification tests, hot functional qualification tests, and periodic in-service operational tests are performed in situ to verify and provide reasonable assurance of the functional ability of the valves. These tests provide reasonable assurance of the reliability of the valve for the design life of the plant.

Stress limits for active and non-active valves are shown in Table 3.9-9 and Table 3.9-8, respectively.

3.9.3.3.4 Non-NSSS Active ASME Code Class 2 and 3 Pumps and Class 1, 2, and 3 Valves

3.9.3.3.4.1 Pumps

ASME Class 2 and 3 safety-related active pumps are listed in Table 3.9-18. The following criteria are employed in a qualification program to provide reasonable assurance of functional capability of the pumps required to function during and following design basis events.

a. Tests or a combination of analyses and tests are used in accordance with ASME QME-1-2007 as accepted in NRC RG 1.100 to confirm that pumps function adequately over the expected range of service conditions, including design basis events and post-design-basis event conditions, as well as inservice inspections and test conditions.
b. The loads imposed by the attached piping, along with the sustained dynamic and seismic loads, are taken into account. The design specification includes applicable loading combinations, and design stress limits for the pumps. To provide reasonable assurance of functional capability under combined loadings, the stresses resulting from the applied test loads envelop the specified service stress limit for which pump functional capability is intended. Design stress limits applied in evaluating loading combinations are described in Subsection 3.9.3.1.3.

3.9.3.3.4.2 Valves

Reasonable assurance of the functional capability of active valves is provided to perform a safety-related function during and after the specified plant design basis events. The active valves are seismically and functionally qualified in accordance with ASME QME-1-2007 as accepted in NRC RG 1.100, and IEEE Std. 344, and are described in Subsection 3.9.6. Subsection 3.9.6 also provides a description of the functional design and qualification provisions and IST programs for safety-related valves.

To provide reasonable assurance of the functional capability of the valves for the plant design life, they are analyzed and/or tested and inspected during construction including the factory tests.

The valves are designed using either stress analysis or conformance with the standard design rules for minimum wall thickness requirements in accordance with the applicable ASME Section III design requirement.

The maximum stress limits used in the analyses are those required by the applicable ASME Section III for the valve that is analyzed. For active valves with extended topworks (e.g., the operator), an analysis of the extended topworks is performed by applying equivalent static seismic loads applied at the center of gravity of the extended topworks.

Prior to installation, functionality and structural integrity tests of the valve are performed as follows:

a. Shell hydrostatic and valve closure tests in accordance with the ASME Section III, NB/NC/ND-3530 requirements

b. Functional and operational tests to verify that the valve opens and closes as required when subjected to the design differential pressure and flow
c. Functional capability qualification of motor operators for seismic and environmental conditions, if any, in accordance with IEEE Std. 323, 344, and 382 (References 53, 35, and 55)

After installation, the following tests are performed to verify and provide reasonable assurance of the functionality of the valve (References 57, 58, 59, and 60). These tests enhance the reliability of the valve for the design life of the plant.

a. Cold hydrostatic tests

b. Hot functional tests

c. Periodic inservice inspections

d. Periodic inservice operations tests

In addition to the valve qualifications noted above, a representative sample of the valves according to type, load level, and size is tested for functional capability during a simulated plant design basis event of an SSE. The seismic qualification of the valves is performed for the SSE preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on Appendix D of IEEE Std. 344.

Selection of damping values for valves to be qualified is determined in accordance with NRC RG 1.61 and IEEE Std. 344. The valve is mounted to represent the typical valve installation or the specified valve installation. The valve includes operators and all appurtenances that are normally attached to the valve in the plant service. Section 3.10 provides the details of seismic qualification.

The functional capability of the active valves with extended topworks during a Level D service (SSE) condition is verified by satisfying the following criteria:

a. The functional capability evaluation of the valve is performed by analysis and/or test when subjected to equivalent static load resulting from the accelerations applied at the center of gravity during Level D service conditions.

b. For evaluation by test, the valve is subjected to a static deflection. The valve is operable within the specified operating time limits while an equivalent static load is applied to the extended structure center of gravity in the direction of the weakest axis of the yoke under design pressure.
c. For evaluation by analysis, stresses at critical locations are evaluated and compared to the allowable stress to determine structural integrity, and deflections at critical locations are compared to evaluate functional capability.

d. Electrical motor operators and other electrical appurtenances necessary for operation are qualified in accordance with IEEE Std. 382 and IEEE Std. 344.

The active valves without extended topworks, such as check valves and other compact valves, are simple in the valve design, and typically there are no masses causing significant distortions in the affection of the valve operation. Therefore, if these valves are designed so that if structural integrity is maintained, reasonable assurance of valve functional capability is considered to be provided.

In addition to the above design considerations, the valves are tested and inspected for in-shop hydrostatic test, in-shop seat leakage test, and periodic valve testing and inspection as installed.

Using the methods as described above, the active valves in the piping system are qualified for the valve functional capability during and after an SSE. In addition, these methods conservatively simulate the seismic event and verify that the active valves will perform their safety-related function when necessary.

3.9.3.4 Component Supports

Jurisdictional boundaries between ASME Section III, Class 1, 2, and 3 component supports and the building structure are established in accordance with ASME Section III, NF.

ASME Section III, Class 1, 2, and 3 component supports are designed and constructed in accordance with ASME Section III and ASME Code Case(s) approved in NRC RG 1.84.

Seismic Category I component supports are designed to meet the requirements of ASME Section III, NF. Welding fabrication and installation, nondestructive examination (NDE), and acceptance standards are in accordance with ASME Section III, NF, NRC RG 1.124 (Reference 62) and NRC RG 1.130 (Reference 63).

Component support building structures are designed to meet the criteria in Appendix 3.8A.

Component supports that are loaded during normal operation, seismic and following a pipe break (branch line breaks not eliminated by a leak-before-break) are specified for design for
loading combinations of Subsection 3.9.3.1. Design stress limits applied in evaluating
loading combinations for Level A, B, C or D plant conditions described in Subsection
3.9.3.1 are in accordance with ASME Section III. Loads in compression members are
limited to 2/3 of the critical buckling load.

Concrete expansion anchors meet the requirements of ACI-349 and IE Bulletin 79-02 with
the provisions identified in Subsection 3.8.4.5.


Where required, snubber supports are used as shock arrestors for safety-related systems and
components. Snubbers are used as structural supports during a dynamic event such as an
earthquake or a pipe break but during normal operation act as passive devices that
accommodate normal expansions and contractions of the systems without resistance. For
the APR1400, snubbers are minimized to the extent practical through the use of design
optimization.

The snubber is modeled as a linear spring element whose spring rate is determined using
test results between the displacements and the test loads. The displacements include the
effects of end fitting clearances, lost motion and compression of the fluid as well.
Snubber reaction loads are determined from the system analysis with the spring elements.
The snubber end fitting clearance, mismatch of end fitting clearances and mismatch of
activation are minimized especially in multiple snubber application at the same support to
share loads evenly.

Reasonable assurance of snubber operability is provided by incorporating analytical, design,
installation, in-service, and verification criteria. The elements used to provide reasonable
assurance of snubber operability for the APR1400 include:

a. Consideration of load cycles and travel that each snubber undergoes during normal
   plant operating conditions

b. Verification that the thermal growth rates of the system do not exceed the required
   lock-up velocity of the snubber

c. Accurate characterization of snubber mechanical properties in the structural
   analysis of the snubber-supported system
d. For engineered, large-bore snubbers, issuance of a design specification to the snubber supplier describing the required structural and mechanical performance of the snubber and verification that the specified design and fabrication requirements are met.

e. Verification that snubbers are properly installed and operable prior to plant operation through visual inspection and measurement of thermal movements of snubber-supported systems during startup tests.

f. A snubber in-service inspection and testing program, which includes periodic maintenance and visual inspection, inspection following a faulted event, a functional testing program, and repair or replacement of snubbers failing inspection or test acceptance criteria. The inservice testing program for snubbers is described in Subsection 3.9.6.4.

Site-specific information includes a list of all safety-related components that use snubbers in accordance with SRP 3.9.3.

3.9.4 Control Element Drive Mechanisms

The CEDM is designed, constructed, and tested to meet the requirements of GDC 1, 2, 4, 14, 26, 27, and 29, as well as 10 CFR 50.55a.

3.9.4.1 Descriptive Information of Control Element Drive Mechanism

The control element drive mechanism (CEDM) is a magnetic jack-type driving apparatus used to vertically position the control element assemblies (CEAs) as an independent reactivity control system. Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 3.8 m (150 in.) stroke in response to operation signals. The CEDM is capable of operating at the maximum speed of 76.2 cm/min (30 in/min) with minimum and maximum weight of the ESA and CEA, 100 kg (220 lbs) and 204 kg (450 lbs), respectively. The CEDM is capable of exerting 600 lbs minimum up force on the CEA and ESA.

The CEDM is designed to function during and after all plant transients. The CEA drop time for 90 percent insertion is 4.0 seconds maximum. The drop time is defined as the interval between the time the power is removed from the CEDM coils and the time the CEA has reached 90 percent of its fully inserted position. Drop motion begins within
0.5 second after the removal of power. The CEDM is designed to function normally during and after being subjected to seismic loads. The vibratory motion of the SSE is included in the fatigue evaluation in accordance with Subsection 3.9.2.2.3. The CEDM allows for tripping of the CEA during and after an SSE.

The design and construction of the CEDM pressure housing fulfills the requirements of ASME Section III, NB. The pressure housings are capable of withstanding, throughout the design life, all normal operating loads, including the steady-state and transient operating conditions specified for the vessel. Mechanical excitations are also defined and included as a normal operating load. Design condition of the CEDM pressure housings is 17.2 MPa (2,500 psia) at 343.3 °C (650 °F), and normal operating condition is 15.5 MPa (2,250 psia) at 323.9 °C (615 °F). The loading combinations and stress limit categories are presented in Table 3.9-11 and are consistent with those defined in the ASME Section III.

The design duty requirement for the CEDM is a total cumulative CEA travel of 30,480 m (100,000 ft) operation without loss of function and a total number of full-height CEA drops of 1,000. Design duty requirement of 100,000 ft of travel was determined by operational experience taking into account 40-year life time of the active components such as the motor assembly and extension shaft assembly.

The test programs performed in support of the CEDM design are described in Subsection 3.9.4.4.

3.9.4.1.1 Control Element Drive Mechanism Design Description

The CEDMs are mounted on nozzles on the top of the reactor vessel closure head. A CEDM consists of upper pressure housing, motor housing, motor assembly, coil stack assembly, two reed switch position transmitter (RSPT) assemblies, and an extension shaft assembly (ESA). The CEDM is shown in Figure 3.9-7. The drive power is supplied by the coil stack assembly, which is positioned around the motor housing. Two RSPT assemblies are supported by the upper shroud which encloses the upper pressure housing assembly.

The lifting operation consists of a series of magnetically operated step movements. Two sets of mechanical latches are used to engage an ESA. The magnetic force is obtained from the coil stack assembly mounted on the outside of the motor housing.
The CEDM control system actuates the stepping cycle and moves the CEA by a withdrawal or insertion stepping sequence. CEDM-hold is obtained by energizing a latch coil at a reduced current, while all other coils are de-energized. The CEAs are tripped upon interruption of electrical power to all coils. Each CEDM is connected to the CEAs by an ESA.

The axial position of a CEA in the core is indicated by three independent readout systems. One system counts the CEDM steps electronically, and the other two consist of magnetically actuated reed switches located at regular intervals along the upper pressure housing.

3.9.4.1.1.1 Control Element Drive Mechanism Pressure Housing

The CEDM pressure housing consists of the motor housing assembly and the upper pressure housing assembly. The motor housing assembly is attached to the reactor vessel closure head nozzle by means of a threaded joint and seal welding. The upper pressure housing is threaded into the top of the motor housing assembly and seal welded. The upper pressure housing encloses the ESA and contains a vent.

3.9.4.1.1.2 Motor Assembly

The motor assembly is an integral unit that fits into the motor housing and provides the linear motion to the CEA. The motor assembly consists of a latch guide tube, upper latches, and lower latches. Clearances in the motor assembly enable the CEDM to avoid stuck rod condition, which is verified by the tests described in Subsection 3.9.4.4.

Both upper latches and lower latches are used to perform the stepping of the CEA. The upper latch also performs the holding function when CEA motion is not required. Engagement of the ESA occurs when the appropriate set of magnetic coils is energized. Total CEA motion per cycle is 19.1 mm (3/4 in.).

3.9.4.1.1.3 Coil Stack Assembly

The coil stack assembly consists of four large DC magnetic coils mounted on the outside of the motor housing assembly. The coils supply magnetic force to actuate magnets for engaging and driving the ESA. Power for the magnetic coils is supplied from two separate supplies.
A conduit assembly containing the lead wires for the coil stack assembly is located at the side of the upper shroud.

3.9.4.1.4 **RSPT Assembly**

Two RSPT assemblies provide separate means for transmitting CEA position indication. Reed switches and voltage divider networks are used to provide two independent output voltages proportional to the CEA position. The RSPT assemblies are positioned to use the permanent magnet in the top of the ESA. The permanent magnet actuates the reed switches as it is passed by them.

3.9.4.1.5 **ESA**

The ESAs are used to link the CEDMs to the CEAs. The ESA has a permanent magnet assembly at the top for actuating reed switches in the RSPT assemblies. The center section of the ESA is called the drive shaft, and the lower end of it is a coupling device for connection to the CEA.

The drive shaft is threaded and pinned to the extension shaft. The drive shaft has circumferential notches in 19.1 mm (3/4 in.) increments along the shaft to provide the means of engagement to the motor assembly.

3.9.4.1.2 **Description of the CEDM Motor Operation**

Withdrawal or insertion of the CEA is accomplished by programmed electric current to the magnetic coils. There are three programmed conditions for each magnetic coil; high current for initial gap closure, low current for maintaining the gap closed, and zero current to allow opening of the gap.

3.9.4.1.2.1 **Operating Sequence for the Double Stepping Mechanism**

The initial condition is the hold mode. In this condition, the upper latch coil is energized with low current.

a. Withdrawal

1) The upper lift coil is energized, causing the 11.1 mm (7/16 in.) upper lift gap to close lifting the CEA.
2) The lower latch coil is energized, causing the lower latches to engage the drive shaft with 0.8 mm (1/32 in.) clearance.

3) The upper lift coil is de-energized, allowing the upper latches to drop 19.1 mm (7/16 in.) and the drive shaft to lower 0.8 mm (1/32 in.), placing the load on the lower latches.

4) The upper latch coil is de-energized, disengaging the upper latches.

5) The lower lift coil is energized, lifting the drive shaft 9.5 mm (3/8 in.).

6) The upper latch coil is energized, engaging the upper latches in the drive shaft with 0.8 mm (1/32 in.) clearance.

7) The lower lift coil is de-energized, allowing the lower latches to drop 9.5 mm (3/8 in.) and causing the drive shaft to drop 0.8 mm (1/32 in.), applying the load on the upper latches.

8) The lower latch coil is de-energized, disengaging the lower latches from the drive shaft.

b. Insertion

1) The lower latch coil is energized, causing the lower latches to engage the drive shaft with 8.7 mm (11/32 in.) clearance.

2) The lower lift coil is energized, lifting the lower latches 9.5 mm (3/8 in.) and lifting the drive shaft 0.8 mm (1/32 in.) thus applying the load to the lower latches.

3) The upper latch coil is de-energized, causing the upper latches to disengage the drive shaft.

4) The upper lift coil is energized, moving the de-energized upper latch assembly up 11.1 mm (7/16 in.).

5) The upper latch coil is energized, engaging the latches with 8.7 mm (11/32 in.) clearance.
6) The lower lift coil is de-energized, allowing the lower latch to drop 9.5 mm (3/8 in.). The drive shaft moves down 8.7 mm (11/32 in.), stopping on the upper latch assembly, which is energized and in its up position.

7) The lower latch coil is de-energized, disengaging the lower latches.

8) The upper lift coil is de-energized, lowering the upper latch assembly with the drive shaft 11.1 mm (7/16 in.).

3.9.4.2 Applicable CEDM Design Specifications

The quality assurance requirements of ASME Section III, NCA, and ASME NQA are satisfied for design, fabrication, and test of the CEDM. Classification of the CEDM components is provided in Table 3.2-1.

The components forming the pressure boundary are the motor housing assembly, upper pressure housing assembly, vent stem, and housing nut. The pressure boundary components are designed, constructed, and tested in accordance with ASME Section III, NB, to provide reasonable assurance of extremely low probability of leakage or gross rupture. The material of the CEDM is described in Subsection 4.5.1.

The adequacy of the design of the non-pressure boundary components has been verified by the life tests as described in Subsection 3.9.4.4.

The RSPT assembly is designed to conform with IEEE Std. 323-2003 (Reaffirmed 2008) and IEEE Std. 344-2004 (Reaffirmed 2009). The electrical components are external to the pressure boundary and are non-pressurized.

3.9.4.3 Design Loads, Stress Limits and Allowable Deformations

The CEDM stress analyses consider the following loads:

a. Reactor coolant pressure and temperature

b. Reactor operating transient conditions

c. Normal operating loads

1) Deadweight
2) Impulse load due to stepping of the CEDM

3) Mechanical base excitation loads

4) Loads produced by the thermal expansion of the reactor vessel closure head

d. IRWST discharge loads
e. BLPB loads
f. Seismic loads

The design and fabrication of the CEDM pressure boundary components fulfill the requirements of ASME Section III, NB. The pressure housings are capable of withstanding all the steady-state and transient operating conditions specified in Table 3.9-11 for a 60-year life. The design report for the ASME Code Class 1 components is to be prepared in accordance with ASME Section III.

Deformation of the CEDM under seismic conditions is evaluated to verify scramability as presented in Subsection 3.9.2.7.3.

The adequacy of the design of the CEDM pressure boundary and non-pressure boundary components has been verified by the life cycle tests as described in Subsection 3.9.4.4.

3.9.4.4 CEDM Operability Assurance Program

The APR1400 CEDM is essentially identical to the System 80 CEDM, which is presently operating at the Palo Verde Nuclear Generating Station, except for the material of the motor housing lower end fitting and thickness of the upper shroud tube. The material of the motor housing lower end fitting was changed to Alloy 690TT from Alloy 600 to improve structural integrity against PWSCC, and thickness of the upper shroud was increased to improve mechanical strength. Besides the design changes of the CEDM, there have been design changes of interfacing components. Material of the RV head nozzle was changed to Ally 690TT from Alloy 600, and the outside diameter of longer RV head nozzle (CEDM nozzle) was increased. Seismic support was installed at the upper portion of the CEDM to restrain horizontal deflection. These changes enhance the structural integrity of the CEDM and do not affect the safety-related functions of the APR1400 CEDM. The following describes the tests performed during development of System 80 CEDM, which provides design verification for the APR1400 CEDM.
For the life cycle test, the CEDM was installed on a test facility that was operated at a nominal temperature of 315.6 °C (600 °F) and a gauge pressure of 15.5 MPa (2,250 psig). The CEDM was operated for a total travel length of 47,854 m (157,000 ft) with no abnormality, which is about 1.5 times the design duty requirement.

During the CEA scram test, 300 full-height drops were completed. All release times were less than 0.3 second, and CEA drop times to 90 percent of full insertion were less than 4.0 seconds, which meets the design criterion.

The life cycle test and scram test provide verification of the ability to overcome a stuck rod because the CEDM operated properly without a stuck rod during such severe tests.

Operating experience also provides design verification of the APR1400 CEDM. The APR1400 CEDM is essentially identical to the CEDM of Palo Verde, HBN 3&4, HBN 5&6, HUN 3&4, HUN 5&6, SKN 1&2, and SWN 1, which are all in operation. The experience has demonstrated that the CEDM operates without malfunction.

First production test programs were completed on the CEDM to verify operability. During the course of this program, more than 1,219 m (4,000 ft) of travel was accumulated and 30 full-height gravity drops were made without mechanism malfunction or measurable wear on operating parts. The program included the following:

a. Operation at 76.2 cm/min (30 in/min) traveling 159 kg (350 lb) of weight at ambient temperature and a gauge pressure of 0.7 MPa (100 psig) for 15.2 m (50 ft)

b. Fifteen full-height drops at simulated reactor operating conditions with 159 kg (350 lb) of weight during the first 61 m (200 ft) travel at 76.2 cm/min (30 in/min)

c. Fifteen full-height drops at simulated reactor operating conditions with 159 kg (350 lb) of weight after traveling 1,162 m (3,812 ft)

d. Operation at simulated reactor operating conditions traveling 159 kg (350 lb) of weight for over 1,219 m (4,000 ft)

e. Operation at 76.2 cm/min (30 in/min) traveling 159 kg (350 lb) at ambient temperature and a gauge pressure of 0.7 MPa (100 psig) for 15.5 m (51 ft)

The mechanism operated without malfunction throughout the test program and, upon final inspection, no measurable wear was found.
Production testing is performed prior to shipment to confirm the capability of the CEDM to meet operation requirements. All production CEDM motors are tested at both ambient pressure and fluid temperatures and at simulated reactor operating conditions for a minimum of 122 m (400 ft) of travel and six full-height gravity drop tests at simulated reactor operating conditions.

No malfunction such as mis-stepping is allowed during the tests, and the total drop time is required to be less than 4.0 seconds with a release time of less than 0.5 second.

After installation of the CEDMs, initial startup testing is performed to verify the insertion, withdrawal, and drop time. The initial startup test program is described in Section 14.2.

3.9.5 Reactor Pressure Vessel Internals

Reactor pressure vessel internals described as reactor internals in this subsection refer to the core support structures and internal structures. Core support structures are those structures or parts of structures which are designed to provide direct support or restraint of the core within the reactor vessel. Internal structures are all structures within the pressure vessel other than core support structures, fuels, control element assemblies, and instrumentation.

3.9.5.1 Design Arrangements

The components of the reactor internals are divided into two major parts consisting of the core support barrel assembly and the upper guide structure assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the reactor internals and is affixed to the bottom head of the reactor vessel. The arrangement of these components is shown in Figure 3.9-8. The flow paths of the main coolant flow and bypass flow within the reactor vessel are described in Subsection 4.4.2.6.1.

The component classification for core support structures and internal structures as reactor internals is summarized below:

a. Core support structures:
   1) Core support barrel
   2) Lower support structure
   3) Upper guide structure barrel assembly
b. Internal structures:

1) Core shroud
2) Alignment keys
3) Hold-down ring
4) Core support barrel snubbers
5) In-core instrumentation nozzle assembly
6) Core support barrel outlet nozzles
7) Inner barrel assembly
8) Guide lugs
9) Heated junction thermocouple tube assembly

The design arrangements including functional requirements for each component of the reactor internals are described in detail below.

3.9.5.1.1 Core Support Barrel Assembly

The major structural member of the reactor internals is the core support barrel assembly. The core support barrel assembly consists of the core support barrel, the lower support structure and in-core instrumentation nozzle assembly, and the core shroud. The material for the assembly is austenitic stainless steel.

The core support barrel assembly is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keyways in the reactor vessel ledge and closure head. The lower flange of the core support barrel supports, secures, and positions the lower support structure and is attached to the lower support structure by means of a welded flexural connection. The lower support structure provides support for the core by means of support beams that transmit the load to the core support barrel lower flange. The locating pins in the beams provide orientation for the lower ends of the fuel assemblies. The core shroud provides a flow path for the coolant and limits the amount of coolant bypass flow. Support and positioning for the fuel
assemblies are provided by the lower support structure. The lower end of the core support barrel is restricted from excessive lateral and torsional movement by six snubbers that interface with the reactor vessel wall. The core support barrel assembly is shown in Figure 3.9-9.

3.9.5.1.1.1 Core Support Barrel

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the reactor vessel. The core support barrel supports the lower support structure upon which the fuel assemblies rest. Shrink-fit into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide alignment between these components in the reactor vessel flange region.

The upper section of the core support barrel contains two outlet nozzles that interface with internal projections on the reactor vessel outlet nozzles to minimize leakage of coolant from inlet to outlet. Since the weight of the core support barrel is supported at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally spaced lugs around the circumference of the core support barrel and act as a tongue-and-groove assembly with the mating lugs on the reactor vessel. Minimizing the clearance between the tongue-and-groove assembly limits the amplitude of vibration. During assembly, as the reactor internals are lowered into the reactor vessel, the reactor vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansions of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The reactor vessel lugs have bolted and captured nickel-based alloy X-750 shims. The core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The reactor vessel shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown in Figure 3.9-10.

3.9.5.1.1.2 Lower Support Structure and ICI Nozzle Assembly

The lower support structure and in-core instrumentation (ICI) nozzle assembly position and support the fuel assemblies, core shroud, and ICI nozzles. The structure is a welded
assembly consisting of a short cylinder, support beams, a bottom plate, ICI nozzles, and ICI nozzle support plate. The lower support structure is made up of a short cylindrical section enclosing an assemblage of grid beams arranged in egg-crate fashion. The outer ends of these beams are welded to the cylinder. Fuel assembly locating pins are attached to the top of the beams. The bottoms of the main support beams in one direction are welded to an array of plates which contain flow holes to provide proper flow distribution. These plates also provide support for the ICI nozzles, support columns, and ICI nozzle support plate. The cylinder guides the main coolant flow and limits the core shroud bypass flow by means of holes located near the base of the cylinder. The ICI nozzle support plate provides lateral support for the ICI nozzles. This plate is provided with flow holes for the requisite flow distribution. The lower support structure and ICI nozzle assembly are shown in Figure 3.9-11.

3.9.5.1.1.3 Core Shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The core shroud consists of a welded vertical assembly of plates designed to channel the coolant through the core. Circumferential rings and top and bottom end plates provide lateral support. The rings are attached to the vertical plates by means of full-length welded ribs and horizontal braces. A small gap is provided between the core shroud outer perimeter and the core support barrel in order to provide upward coolant flow in the annulus, thereby minimizing thermal stresses in the core shroud. The core shroud is shown in Figure 3.9-12. Four hard-faced guide lugs, spaced 90 degrees apart, protrude vertically from the top of the core shroud and engage in corresponding hard-faced slots in the upper guide structure fuel alignment plate to provide reasonable assurance of proper alignment between the upper guide structure assembly and core shroud/lower support structure.

3.9.5.1.2 Upper Guide Structure Assembly

The upper guide structure (UGS) assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the control element spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition and protects the control elements from the effects of coolant cross flow in the upper plenum. The UGS assembly is handled as one unit during installation and refueling.
The UGS assembly consists of the UGS barrel assembly and the inner barrel assembly (IBA) (Figure 3.9-13). The UGS barrel assembly consists of UGS support barrel, fuel alignment plate, UGS support plate, and control element guide tubes. The UGS support barrel consists of a right circular cylinder welded to a ring flange at the upper end and to a circular plate (UGS support plate) at the lower end. The flange, which is the supporting member for the entire UGS assembly, seats on its upper side against the reactor vessel head during operation. The lower side of the flange is supported by the hold-down ring, which rests on the core support barrel upper flange. The UGS flange and the hold-down ring engage the core support barrel alignment keys by means of four accurately machined and located keyways equally spaced at 90-degree intervals. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the control element drive mechanisms. The fuel alignment plate is positioned below the UGS support plate by cylindrical control element guide tubes. These tubes are attached to the UGS support plate and the fuel alignment plate by rolling the tubes into the holes in the plates and welding. The fuel alignment plate is designed to align the lower ends of the control element guide tubes which in turn locate the upper ends of the fuel assemblies. The fuel alignment plate also has four equally spaced slots on its outer edge that engage with Stellite hard-faced lugs protruding from the core shroud to provide alignment. The control element guide tubes bear the upward force on the fuel assembly hold-down devices. This force is transmitted from the fuel alignment plate through the control element guide tubes to the UGS barrel support plate.

The IBA limits crossflow and provides separation of the CEA. The IBA consists of top plate welded to a right circular barrel open at the bottom and containing an assemblage of large vertical tubes connected by vertical plates in a grid pattern welded to the inside of the barrel. The IBA is held in position by continuous weld between the barrel flange and the top surface of the UGS barrel upper flange.

Guides for the CEA extension shafts are provided by the top plate of the IBA. The tubes and connecting plates within IBA are furnished with multiple holes to permit hydraulic communication.

The hold-down ring provides axial force on the flanges of the UGS assembly and the core support barrel assembly in order to prevent movement of the structures under hydraulic forces. The hold-down ring is designed to accommodate the differential thermal expansion between the reactor vessel and the reactor internals in the vessel ledge region.
3.9.5.1.3 Flow Skirt

The nickel-based alloy flow skirt is a right circular cylinder, perforated with flow holes, and reinforced with two stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The flow skirt is supported by nine equally spaced machined sections that are welded to the bottom head of the reactor vessel.

3.9.5.1.4 In-Core Instrumentation Support System

The in-core neutron flux monitoring system includes self-powered, in-core detector assemblies, supporting structures and guide paths and an amplifier system to process detector signals. The self-powered in-core detector assemblies and the amplifier system are described in Section 7.7. The instrumentation supporting structures and guide paths are described in this section and shown in Figure 3.9-14.

The support system begins outside the reactor vessel, penetrates the bottom of the vessel boundary and terminates in the upper end of the fuel assembly. Each in-core instrument is guided over the full length by the external guidance conduit, ICI guide tube nozzles of the reactor vessel, the lower support structure, and the guidepost of the fuel assembly. Figure 3.9-14 shows the in-core instrument support structure. The in-core instrumentation support system routes the instruments so that detectors are located in selected fuel assemblies throughout the core. An equal instrument has the same length for all locations. The guide tube routing outside the reactor vessel is a simple 180-degree bend to the seal table. The pressure boundaries for the individual instruments are at the out-of-reactor seal table, where the external electrical connections to the in-core instruments are made. Each instrument has an integral seal plug which forms a seal at the instrument seal table and through which the signal cables pass. Static O-ring seals are used to seal against operating pressure.

The static O-ring seals are classified as safety-related, and shall be capable of leak tight operation for a design life of two years under the following design conditions:

a. Pressure difference from 0 to 2,500 psi.

b. Temperature range from 40 °F to 150 °F.

c. Radiation level of 2E04 Gy (This value includes gamma and neutron radiation).
3.9.5.2 Loading Conditions

The following loading conditions are considered in the design of the reactor internals:

a. Normal operating temperature differences
b. Normal operating pressure differences
c. Flow loads
d. Weights, reactions, and superimposed loads
e. Vibration loads
f. Shock loads (including SSEs)
g. Anticipated transient loadings
h. Handling loads (not combined with other loads above)
i. Secondary side break, and LOCA loads
j. IRWST discharge loads

3.9.5.2.1 Design Loadings

The following loading combination is considered as a design loading.

Normal operation loads in combination with IRWST discharge loads. Normal operation loads are defined as the following sustained loads resulting from the normal events:

a. Pressure difference
b. Temperature
c. Mechanical loads
   1) Weight
   2) Loads from flow impingement or flow of reactor coolant
   3) Superimposed or reaction loads
3.9.5.2.2 Level A Service Loadings

The following loading combination is considered as Level A service loadings.

Normal operation loads in combination with specified system operating transient loads resulting from normal events.

3.9.5.2.3 Level B Service Loadings

The following separate loading combinations are considered as Level B service loadings.

   a. Normal operation loads in combination with IRWST discharge loads and system operating transient loads from the upset events. The IRWST discharge loads are defined as the loads due to postulated discharge to in-containment refueling water storage tank.

   b. Normal operation loads in combination with the system operating transient loads from the upset event (the loss of external load with turbine control system failure). Note that the loss of external load of the upset event, which is evaluated as if it occurs once during the plant lifetime, is the emergency event. This event is evaluated with this combination of loadings for conservatism.

3.9.5.2.4 Level C Service Loadings

There are no Level C service loadings (Refer to Subsection 3.9.3.1).

3.9.5.2.5 Level D Service Loadings

The following loading combination is considered as Level D service loadings.

   a. Normal operation loads

   b. Either the main steam/feed water pipe break (MS/FWPB) or LOCA loads (including asymmetric blowdown loads), whichever are greater

   c. SSE loads

   d. IRWST discharge loads
LOCA is defined as the loss of reactor coolant at a rate in excess of the reactor coolant normal makeup rate, from breaks in the reactor coolant pressure boundary inside primary containment up to, and including, a break equivalent in size to the largest primary branch line not eliminated by leak-before-break (LBB) criteria.

3.9.5.3  Design Bases for Reactor Internals

The RCS transient design basis for reactor internals is addressed in Subsection 3.9.1.1.

The potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on reactor internals are addressed in Subsection 3.9.2.3. The CVAP for reactor internals is addressed in Subsection 3.9.2.4. The dynamic system analysis of reactor internals under faulted conditions is addressed in Subsection 3.9.2.5.

The reactor internals are designed to meet interface cold gaps between reactor internals and the reactor vessel and between the main parts of the reactor internals.

The reactor internals are designed, fabricated, erected, and tested to conform with the requirements of 10 CFR 50.55a, 10 CFR 52.47(b)(1), 10 CFR 52.80(a), and 10 CFR Part 50, Appendix A (GDC 1, 2, 4, and 10).

The design code, code cases, and acceptance criteria applicable to the design, analysis, fabrication, and nondestructive examination are provided in the design specification to meet the structural and functional integrity of the reactor internals in accordance with ASME Section III. The design specification and design report are prepared in accordance with the NCA requirements of ASME Section III.

The stress limits to which the reactor internals are designed are listed in Table 3.9-12.

The loading categories and stress limits are defined in the applicable section of ASME Section III. The design and construction of core support structures and internal structures are in accordance with ASME Section III as described in Table 3.9-12.

To properly perform their functions, the reactor internals are designed to meet the following deformation limits:

a. Under Level A, Level B, and Level C service loadings, the core is held in place, and deflections are limited so that the CEAs can be inserted under their own weight as the only driving force.
b. Under Level D service loadings that require CEA insertability, deflections are limited so that the core is held in place, adequate core cooling is preserved, and all CEAs can be inserted. Those deflections that would influence CEA movement are limited to less than 80 percent of the deflections required to prevent CEA insertion.

The allowable deformation limits are established as 80 percent of the loss-of-function deflection limits.

The significant component deflection limits are designed as follows:

1) Relative displacement of fuel lower end fitting interface with the lower support structure avoids disengagement.

2) Relative displacement of fuel upper end fitting interface with the upper guide structure avoids disengagement.

3) The CEA guide tube lateral deflection allows CEA insertion.

In the design of critical reactor internals that are subject to fatigue, stress analysis is performed using the design fatigue curve of Figure I-9.2 of ASME Section III. A cumulative usage factor of less than one is used as the limiting criterion.

As indicated in the preceding sections, the stress and fatigue limits for reactor internals are obtained from ASME Code. Allowable deformation limits are established as 80 percent of the loss-of-function deflection limits. These limits provide adequate safety factors providing reasonable assurance that as long as calculated stresses, cumulative usage factors, or deformations do not exceed these limits, the design is conservative.

A summary of the maximum calculated total stress, deformation, and cumulative usage factor for each service limit of core support structures in accordance with ASME Section III is provided.

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

This section describes the functional design and qualification provisions and inservice testing (IST) programs for safety-related pumps, valves, and dynamic restraints. The qualification provisions and IST programs are described to verify that these components are
in a state of operational readiness to perform their safety functions throughout the life of the plant.

The APR 1400 utilizes an IST program for ASME Section III, Class 1, 2 and 3 safety-related pumps, valves, and dynamic restraints. The IST program is developed in accordance with the requirements of ASME OM Code (Reference 33), as required by 10 CFR 50.55a(f) and the acceptable ASME Code Cases listed in NRC RG 1.192 (Reference 64) that are incorporated by 10 CFR 50.55a(b).

ASME OM Code, ISTB applies to the inservice testing requirements for pumps, Subsection ISTC applies to the inservice testing requirements for valves, and Subsection ISTD applies to the inservice testing requirements for dynamic restraints. Additionally, APR 1400 IST program incorporates the guidance and information on the format and content for IST program, and relief requests of NUREG-1482, Rev. 2 (Reference 58).

The sections of 10 CFR Part 50, General Design Criteria (GDC), that apply to this section are:

a. GDC 1 as it relates to structures, systems, and components (SSCs), which include pumps, valves, and dynamic restraints being designed, fabricated, erected, and tested to quality standards commensurate with the safety-related functions to be performed. These requirements provide reasonable assurance that safety-related components and structures meet service loading conditions, stress limits, and quality requirements of ASME Code permitted in 10 CFR 50.55a.

b. GDC 2 as it relates to components important to safety being designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The effects of expected natural phenomena on normal and accident conditions are considered in the loading combinations for components important to safety. Additional design information is provided in the sections that describe the individual safety-related SSCs.

c. GDC 4 as it relates to components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The safety-related SSCs are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-
coolant accidents (LOCA). Additionally, the APR1400 design is based on the leak-before-break (LBB) concept, as described in Section 3.6.3, to eliminate the dynamic effects of postulated pipe rupture.

d. GDC 14 as it relates to the reactor coolant pressure boundary (RCPB) being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The RCPB is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including anticipated transients, with stresses within applicable limits.

e. GDC 15 as it relates to the reactor coolant system (RCS) being designed with sufficient margin to provide reasonable assurance that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOO). Steady-state and transient analyses are performed to ensure that the design conditions of the RCS are not exceeded during normal operation. Additionally, RCPB components have sufficient margin of safety based on the use of proven materials and design codes, the use of proven fabrication techniques, nondestructive shop examination, and integrated hydrostatic testing of assembled components.

f. GDC 37 as it relates to the emergency core cooling system (ECCS) being designed to permit appropriate periodic pressure and functional testing to confirm the structural and leak-tight integrity of its components, and the operability and performance of the active components of the system. The ECCS is provided with testing capability to demonstrate system and component leak-tight integrity, operability and performance.

g. GDC 40 as it relates to the containment heat removal system being designed to permit appropriate periodic pressure and functional testing to confirm the structural and leak-tight integrity of its components, and the operability and performance of the active components of the system. System piping, valves, pumps, and other components of the containment heat removal system are arranged so that each component can be tested periodically for leak-tight integrity and operability.
h. GDC 43 as it relates to the containment atmospheric cleanup system being designed to permit appropriate periodic pressure and functional testing to ensure the structural and leak-tight integrity of its components, and the operability and performance of the active components of the system, including pumps and valves. Testing of the CSS is conducted to ensure structural and leak-tight integrity, and operability and performance in accordance with GDC 40. In addition, performance testing is conducted on active components of the CSS.

i. GDC 46 as it relates to the cooling water system being designed to permit appropriate periodic pressure and functional testing to confirm the structural and leak-tight integrity of its components, and the operability and performance of the active components of the system. The design provides measures for periodic testing of active components of the cooling water systems for operability and functional performance.

j. GDC 54 as it relates to piping systems penetrating the primary reactor containment being provided with leak detection and isolation capabilities to test periodically the operability of the isolation valves and determine valve leakage acceptability. Piping systems that penetrate the containment are designed to provide the required isolation and testing capabilities.

Other Sections that interface with this section are:

a. Subsection 3.2.2 addresses the classification system and quality group for pumps and valves.

b. Subsection 3.9.2 addresses dynamic testing and analysis of safety-related pumps, valves, and snubbers.

c. Subsection 3.9.3 addresses the structural design of safety-related pumps, valves, and snubbers.

d. Section 3.10 addresses the seismic and dynamic qualification of safety-related pumps and valves.

e. Section 3.11 addresses the environmental qualification of safety-related pumps and valves.
f. Section 3.12 addresses the design and leak testing provisions of pressure retaining systems and components that interface with the reactor coolant system as part of the primary review responsibility for intersystem LOCAs.

g. Section 3.13 addresses programs for ensuring the adequacy and integrity of bolting and threaded fastener.

h. Subsection 5.2.2 addresses the valves specified for overpressure protection of the reactor coolant pressure boundary.

i. Sections 5.4.7 and 6.3 address residual heat removal and emergency core cooling systems piping, each of which is connected to the reactor coolant system and is subject to thermally stratified flow, thermal striping, and/or thermal cyclic effects.

j. Subsection 6.2.1.2 addresses the analyses of subcompartment differential pressures resulting from postulated pipe breaks.

k. Subsections 6.2.4 and 6.2.6 address the containment isolation system and the overall containment leakage testing program, respectively.

l. Subsection 9.2.1 addresses surveillance, testing, inspection, and maintenance programs of essential service water systems.

m. Section 10.3 addresses the safety-related portion of the main steam system.

n. Section 14.2 addresses preoperational and initial startup testing for systems that contain safety-related pumps, valves, and dynamic restraints.

o. Section 17.6 describes the program for implementation of the Maintenance Rule for systems that contain safety-related pumps, valves, and dynamic restraints.

ASME OM Code describes the IST scope and establishes the requirements for preservice and inservice testing and examination of certain components to assess their operational readiness. ASME OM Code identifies the components subject to test or examination, responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating the results, corrective action, personnel qualification, and record keeping.

These requirements apply to:
a. Pumps and valves that are required to perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident.

b. Pressure relief devices that protect systems or portions of systems that perform one or more of the three functions identified above.

c. Dynamic restraints used in systems that perform one or more of the three functions identified above, or that ensure the integrity of the RCPB.

The COL applicant is to provide a full description of the IST program (including preservice testing (PST) and MOV testing) for pumps, valves, and dynamic restraints as required by 10 CFR 50.55a, that will be administratively controlled such that the applicable requirements of ASME OM Code edition and addenda are incorporated in the IST program (COL 3.9(3)).

ASME Code, Section III, Class 1, 2, 3, and non-ASME Code safety-related pumps, valves, and dynamic restraints are incorporated into a 10 year interval IST Program.

3.9.6.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

The functional qualification of safety-related pumps, valves, and dynamic restraints (snubbers) are performed in accordance with ASME QME-1-2007 (Reference 51), as endorsed in NRC RG 1.100 (Reference 34).

The functional design and qualification of safety-related pumps, valves, and snubbers includes:

a. Safety-related pump, valve, and piping designs include provisions that allow testing of pumps and valves at the maximum flow specified in the plant accident analyses.

b. The functional design and qualification of each safety-related pump and valve is performed such that each pump and valve are capable of performing its intended function for a full range of system differential pressure and flow, ambient temperatures, and available voltage (as applicable) under conditions ranging from normal operating to design-basis accidents.
c. The APR1400 design provides ready access to SSC to facilitate comprehensive testing using currently available equipment and techniques. Accessibility incorporated into the design complies with the requirements of ASME OM Code and 10 CFR 50.55a(f). System design incorporates provisions, including alternate flow paths and required instrumentation, to allow full flow testing of pumps under the IST program. The design also incorporates provisions to permit ready IST of valves.

d. The provisions for the functional design and qualification of snubbers are provided in Subsections 3.9.3 and 3.9.6.4. Snubbers in safety-related systems include provisions to allow access for IST program activities.

e. The design and installation of safety and relief valves are described in Section 3.9.3.

f. The seismic and dynamic qualification of mechanical and electrical equipment is described in Section 3.10.

g. The environmental qualification of safety-related pumps and valves is described in Section 3.11.

h. Safety-related valves that are part of the RCPB are designed and tested such that these valves will not experience any abnormal leakage, or increase in leakage, from their loading, as addressed in Section 3.10.

i. Pumps, valves, and snubbers are designed with sufficient margin to demonstrate that the design conditions are not exceeded.

j. Pump motors are designed to tolerate anticipated frequency and voltage variations due to degraded electrical power supply line conditions.

Pumps and valves are tested within the qualification program requirements to confirm that the required components are capable of performing their intended safety function. The safety analysis includes information concerning the design limitations and functional requirements for the performance of pumps and valves, including operation of them at the maximum flow rate. The pump functional design and qualification include an assessment for degraded flow conditions. The qualification program requires pump and valve testing over the full range of system differential pressures, flow rates, temperatures, and available
voltages (as applicable), from normal operating to design basis conditions, and considers degraded flow that may occur during post-accident conditions.

3.9.6.2 Inservice Testing Program for Pumps

IST program for safety-related pumps is developed in accordance with the requirements of ASME OM Code, ISTA and ISTB. Pumps subject to IST in accordance with the ASME Code are listed in Table 3.9-13. This table includes the safety class, test parameters, and the frequency at which the testing is performed.

This program includes baseline pre-service testing to support the periodic in-service testing of the safety-related pumps. Preservice testing is performed on pumps prior to initial plant operation. Depending on the test results, the plan will provide commitment to disassemble and inspect the safety-related pumps.

For each size, type, and model of pump, testing that encompasses design conditions is performed to demonstrate acceptable flow rate and corresponding pump head, bearing vibration levels, and pump internals wear rates for the operating time specified for each system mode of pump operation. From these tests, baseline reference value of pump speed, hydraulic performance and vibration data for evaluating the acceptability of the pump after installation are also developed. Test data are used to provide reasonable assurance that the pump specified for each application is not susceptible to inadequate minimum flow rate and inadequate thrust bearing capacity. With respect to minimum pump flow operation, the sizing of each minimum recirculation flow path is evaluated to provide reasonable assurance that its use under all analyzed conditions does not result in degradation of the pump. The flow rate through minimum recirculation flow paths can be measured periodically to verify that flow is in accordance with the design specification.

The methods, range and accuracy of measurements used to measure pressure, flowrate, speed, and vibration meet the requirements of ASME OM Code, ISTB-5000. The range, accuracy, instrument locations, fluctuations, and frequency response range meet the requirements of ISTB-3510, ASME OM Code.

The safety-related pumps and piping configurations accommodate the IST at a flow rate at least as large as the maximum design flow for the pump application. The safety-related pumps are provided with instrumentation to verify that net positive suction head available (NPSHA) is greater than or equal to the net positive suction head required (NPSHR) during all modes of pump operation. These pumps can be disassembled for evaluation when
ASME OM Code, ISTB testing results in a deviation that falls within the required action range. The code provides criteria limits for the test parameters identified in Table 3.9-13. The detailed IST program establishes the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related pumps, including the basis for the frequency and the extent of each disassembly. Factors to be considered in the frequency and extent of disassembly include, but are not limited to the following:

a. Performance history of the pump to identify pumps that are prone to degradation/wear.

b. Analysis of trends of pump test parameters and service conditions.

c. Analysis of pump components that are subject to aging and require an approach to maintenance and replacement (e.g., O-rings).

d. Results of non-intrusive pump testing. The non-intrusive technologies used may obviate the need for inspection/disassembly of safety-related pumps, provided the technologies demonstrate an equivalent ability to detect pump degradation as inspection/disassembly would.

The testing requirements and acceptance criteria are identified in ISTB-5000.

The program may be revised throughout the plant life to minimize disassembly based on disassembly experience.

If ASME OM Code, ISTB pump tests cannot be performed on the component cooling water (CCW) or essential service water (ESW) pumps due to inability to repeat pump test at single point flow conditions, pump curve testing will be used to assess pump degradation in accordance with ASME OM Code Case OMN-16 accepted for use in NRC RG 1.192 that is incorporated by reference in 10 CFR 50.55a. The following provisions shall be complied with in the use of pump curve testing for the CCW/ESW pumps:

a. Pump curves are developed, or manufacturer's pump curves are validated, when the pumps are known to be operating acceptably.

b. The reference points used to develop or validate the curve are measured using instruments at least as accurate as required by the code.

c. Pump curves are based on an adequate number of points, with a minimum of five.
d. Points are beyond the “flat” portion (low flow rates) of the curves in a range which includes or is as close as practicable to design basis flow rates.

e. Acceptance criteria based on the curves does not conflict with Technical Specifications or FSAR design bases.

f. If vibration levels vary significantly over the range of pump conditions, a method for assigning appropriate vibration acceptance criteria should be developed for regions of the pumps curve.

g. When the reference pump curve may have been affected by repair, replacement, or routine service, a new reference curve shall be determined or the previous curve revalidated by an inservice test.

The COL applicant is to provide an IST program including the type of testing and frequency of site-specific pumps subject to IST in accordance with the ASME OM Code and Table 3.9-13 (COL 3.9(4)).

3.9.6.3 **Inservice Testing Program for Valves**

The IST of safety-related valves is addressed in the IST program.

The valves IST program categories are classified based on the safety-related valve functions. The following categories are used in classifying the valve IST categories in accordance with the ASME OM Code.

a. Category A – valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required functions.

b. Category B – valves for which seat leakage in the closed position is inconsequential for fulfillment of the required functions.

c. Category C – valves that are self-actuating in response to some system characteristic, such as pressure (relief valves) or flow direction (check valves), for fulfillment of the required functions.

d. Category D – valves that are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves.
Category A and Category B valves are tested as follows:

a. Full-stroke exercising of valves during operation at power to the positions is required to fulfill their functions. If full-stroke exercising during operation at power is not practical, the testing may be limited to part-stroke exercising during operation at power and full-stroke exercising during cold shutdowns.

b. If valve exercising is not practical during operation at power, the testing may be limited to full-stroke exercising of the valves during cold shutdowns. Valve exercising may be limited to part-stroke during cold shutdowns and full-stroke during refueling outages.

c. Valve exercising is not required if the time period since the previous full-stroke exercise is less than 3 months. During extended shutdowns, valves that are required to perform their intended safety function are exercised every 3 months.

d. Valve exercising during cold shutdown commences within 48 hours of achieving cold shutdown and continues until testing is complete or the plant is ready to return to operation at power.

e. For extended outages, testing need not be commenced in 48 hours, provided all valves required to be tested during cold shutdown will be tested before or as part of plant startup. However, it is not the intent of this subsection to keep the plant in cold shutdown to complete cold shutdown testing.

f. All valve testing required to be performed during a refueling outage is completed before returning the plant to operation at power.

Category C valves testing are addressed in Subsections 3.9.6.3.3 (check valves) and 3.9.6.3.6 (safety and relief valves).

IST for safety-related valves is developed in accordance with the requirements of ASME OM Code, ISTA and ISTC. Table 3.9-13 lists the valves to be included in the IST program as well as the valve type, valve identification number, code class, valve category, valve functions, required tests, and test frequencies. Safety-related valves include the valves that are necessary to provide reasonable assurance of the following:

a. Integrity of the RCPB
b. Capability to achieve safe shutdown of the reactor and keep it in a safe shutdown condition

c. Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures in excess of the guidance of 10 CFR 100.11

Table 3.9-13 also provides explanatory notes/justifications for any code defined testing exceptions. The IST program includes safety-related valve IST details and test schedules and frequencies, in the inspection and testing program. This program includes baseline preservice testing to support the periodic inservice testing of the safety-related valves. Preservice testing is performed on valves prior to initial plant operation. Depending on the test results, the plan will provide commitment to disassemble and inspect the safety-related valves. The primary elements of this plan, including the requirements of Generic Letter (GL) 96-05 (Reference 60) for motor-operated valves (MOVs), are presented in Subsection 3.9.6.3.1.

The specific testing requirements and acceptance criteria are identified in ISTC-5000.

The COL applicant is to provide an IST program including the type of testing and frequency of any site-specific valves subject to IST in accordance with the ASME OM Code and Table 3.9-13 (COL 3.9(5)).

3.9.6.3.1 Inservice Testing Program for Motor-Operated Valves

In addition to the IST program requirements in the ASME OM Code incorporated by reference in 10 CFR 50.55a(f), 10 CFR 50.55a(b)(3)(ii) requires establishment of a program periodically to ensure that the safety-related MOVs continue to be capable of performing their design basis safety functions. GL 96-05 (Reference 60) provides additional guidance for the periodic verification of the design basis capability of MOVs.

Therefore, the IST program for all safety-related MOVs incorporates requirements of ISTC of ASME OM Code, and applicable addenda, as required by 10 CFR 50.55a(f). The IST program incorporates ASME Code Case OMN-1, “Alternative Rules for Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants,” and OMN-11, as accepted by the NRC with conditions in NRC RG 1.192 (Reference 64). The Code Cases listed in Table 2 of NRC RG 1.192 (Reference 64) are conditionally accepted Code Cases, which may be used without request to the NRC provided it is used with any identified limitations or modifications.
The provision listed in Table 2 of NRC RG 1.192 (Reference 64) includes:

a. The adequacy of the diagnostic test interval for each MOV must be evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of OMN-1.

b. When extending exercise test intervals for high risk MOVs beyond a quarterly frequency, ensure that the potential increase in Core Damage Frequency (CDF) and risk associated with the extension is small and consistent with the intent of the Commission’s Safety Goal Policy Statement.

c. When applying risk insights as part of the implementation of OMN-1, MOVs must be categorized according to their safety significance using the methodology described in Code Case OMN-3, with the conditions discussed in this regulatory guide or use other MOV risk ranking methodologies accepted by the NRC on a plant specific or industry-wide basis with the conditions in the applicable safety evaluations.

The IST program for MOVs also incorporates the guidance of NUREG-1482 (Reference 58). The periodic verification program for all safety-related MOVs incorporates the guidance of GL 96-05 (Reference 60) which supersedes GL 89-10 (Reference 59) and its supplements with regards to MOV periodic performance verification. The MOV periodic verification program also implements the recommendations from the Joint Owners Group (JOG) MOV Periodic Verification Program (MPR-2524-A, November 2006, Reference 65).

The IST of MOVs relies on diagnostic techniques that are consistent with the state-of-the-art and that permit an assessment of the performance of the valve under actual loading. Periodic testing per GL 96-05 (Reference 60) is conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design basis conditions. The detailed IST program includes the optimal frequency of this periodic verification. The frequency and test conditions are sufficient to demonstrate continuing design basis and required operating capability. The IST interval between testing to demonstrate continued design basis capability does not exceed 5 years or three refueling outages, whichever is longer. The code provides criteria limits for the test parameters identified in Table 3.9-13 in accordance with ASME OM Code, IST.

Each MOV is tested in the open and closed states under static and maximum achievable preoperational conditions using diagnostic equipment that measures torque and thrust, and
motor parameters to verify correct MOV actuator sizing and control settings. The MOV is tested under various differential pressure and flow conditions up to maximum achievable conditions to determine torque and thrust requirements at design conditions. The parameters and acceptance criteria, which demonstrate fulfillment of the functional performance requirements, are as follows:

a. As required by the safety function, the valve is fully open or fully closed with diagnostic indication of hard-seat contact.

b. The control switch settings provide adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracies, control switch repeatability, load-sensitive behavior, and margin for degradation.

c. The motor output capability at degraded voltage is equal to or exceeds the control switch setting including consideration of diagnostic equipment inaccuracies, control switch repeatability, load-sensitive behavior, and margin for degradation.

d. The maximum torque and thrust achieved by the MOV, including diagnostic equipment inaccuracies and control switch repeatability, do not exceed the allowable structural capability limits for the individual parts of the MOV.

e. The remote position indication testing verifies that proper disk position is indicated in the control room.

f. Stroke time measurements taken during valve opening and closing meet minimum and maximum stroke time requirements.

The detailed IST program establishes the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related MOVs, including the basis for the frequency and extent of each disassembly. Factors to be considered in the frequency and extent of disassembly include, but are not limited to the following:

a. Performance history of the safety-related valves to identify valves that are prone to degradation/wear

b. Analysis of trends of valve test parameters and service condition

c. Analysis of valve components that are subject to aging and require an approach to maintenance and replacement (e.g., O-rings)
d. Results of non-intrusive valve testing. Use of non-intrusive technologies may obviate the need for inspection/disassembly of safety-related valves altogether, provided the technologies demonstrate an equivalent ability to detect valve degradation as inspection/disassembly would.

The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience.

3.9.6.3.2 Inservice Testing Program for Power-Operated Valves Other than Motor-Operated Valves

IST program of safety-related power-operated valves (air-operated, hydraulic-operated, solenoid-operated) is performed in accordance with the requirements of ASME OM Code by reference in 10 CFR 50.55a(f).

Each power-operated valve (POV) is tested in the open and closed directions under static and maximum achievable preoperational conditions using diagnostic equipment that measures or provides information to determine total friction, stroke time, seat load, spring rate, travel under normal pneumatic or hydraulic pressure (as applicable to the type of POV), and minimum pneumatic or hydraulic pressure. The POV is tested under differential pressure and flow conditions up to maximum achievable conditions and perform tests to determine the force requirements at design conditions.

The force requirements to close the valve to the position at which there is diagnostic indication of full valve closure (as required for the safety function of the applicable valves) will be determined. The determination of design force requirements will be made for such parameters as differential pressure, fluid flow, minimum pneumatic or hydraulic pressure, power supply, temperature, and seismic/dynamic effects for POVs which must operate during these transients. The design force requirements will be adjusted for diagnostic equipment inaccuracies.

The total force delivered by the POV under static and dynamic conditions (including diagnostic equipment inaccuracies) will be measured to compare to the allowable structural capability limits for the assembly and individual parts of the POV. Tests will be conducted for proper MCR position indication of the POV.

Solenoid-operated valves (SOVs) are tested using Class 1E electrical power supply voltage and current to verify SOVs are capable of performing their safety functions at design basis.
accident conditions. SOV tests include confirmation of the energized position and fail position when de-energized.

All safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design conditions. The in-service testing of POVs incorporates the use of advanced non-intrusive techniques to periodically assess degradation and performance characteristics of the POVs. The ASME OM Code, ISTC tests are performed, and valves that fail to exhibit the required performance can be disassembled for evaluation.

Periodic verification testing is conducted under adequate differential pressure and flow conditions per the guidance of Regulatory Issue Summary (RIS) 2000-03 (Reference 57), which incorporates the lessons learned from MOV analyses and tests in response to GL 96-05 (Reference 60). Periodic testing allows a justifiable demonstration of continuing POV capability for design basis conditions.

Additional testing is performed as part of the air-operated valve (AOV) periodic verification program, which includes the elements for an AOV periodic verification program as identified in the JOG air-operated valve program. The AOV periodic verification program incorporates the attributes for a successful POV design capability and long-term periodic verification program, as discussed in RIS 2000-03 (Reference 57) by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of AOVs and other POVs included in the IST program. The lessons learned addressed in the AOV program include:

a. Setpoints for AOV are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).

b. Periodic static testing is performed to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or operating experience, periodic dynamic testing will be performed to re-verify the capability of the valve to perform its required safety function.
c. Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.

d. Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References JOG air-operated valve program (Reference 61) and Comments on JOG air-operated valve program (Reference 66), with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.

e. Post-maintenance procedures include appropriate instructions and criteria to demonstrate baseline testing is re-performed as necessary when maintenance on the valve, valve repair, or replacement has the potential to affect valve functional performance.

f. Guidance is included to address lessons learned from other valve programs in procedures and training specific to the AOV program.

g. Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as part of the AOV program.

h. The attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related POVs, such as electro-hydraulic valves, are applied to those other POVs.

i. Safety-related valves are categorized according to their safety significance and risk ranking to be consistent with the provision in RIS 2000-03. The safety-related air operated valves (AOVs) are assigned the highest category according to the Joint Owners Group (JOG) AOV program (including NRC staff comments provided in a letter to the Nuclear Energy Institute, dated October 8, 1999).

The ASME OM Code, ISTC, provides criteria limits for the test parameters identified in Table 3.9-13.

The detailed IST program establishes the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related POVs, including the basis
for the frequency and the extent of each disassembly. Factors to be considered in the frequency and extent of disassembly include but are not limited to the following:

a. Performance history of the safety-related valves to identify valves that are prone to degradation/wear

b. Analysis of trends of valve test parameters and service conditions

c. Analysis of valve components that are subject to aging and require an approach to maintenance and replacement (e.g., O-rings)

d. Results of non-intrusive valve testing. Use of non-intrusive technologies may obviate the need for inspection/disassembly of safety-related valves altogether, provided the technologies demonstrate an equivalent ability to detect valve degradation as inspection/disassembly would.

The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience.

3.9.6.3.3 Inservice Testing Program for Check Valves

All safety-related piping systems incorporate provisions for testing to demonstrate the operability of check valves under design conditions. The IST of check valves incorporates the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The effects of rapid pump starts and stops, as expected for system operating conditions, are considered in the testing. Conditions of reverse flow that may occur during expected system operation are also considered in the testing. Non-intrusive technique includes acoustic, ultrasonic, magnetic, and x-ray technologies, which are used to measure valve operating parameters (e.g., fluid flow, disk position, disk movement, and disk impact forces). The technique is also used for monitoring an upstream pressure or tank level, and for performing a leak test, a system hydrostatic or pressure test, or radiography.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been fulfilled are as follows:

a. During all tests modes that simulate expected system operating conditions, the valve disk fully opens or fully closes as expected based on the direction of the differential pressure across the valve.
b. Leak-tightness of valve when fully closed is within established limits, as applicable.

c. Valve disk positions are determinable without disassembly.

d. Valve testing must verify free disk movement whenever moving to and from the seat.

e. The disk is stable in the open position under normal and other required system operating fluid flow conditions.

f. The valve is correctly sized for the flow conditions specified, i.e., the disk is in full-open position at normal full-flow operating condition.

g. Valve design features, material, and surfaces accommodate non-intrusive diagnostic testing methods available in the industry or as specified.

h. Piping system design features accommodate all the applicable check valve testing requirements as described in Table 3.9-13.

The ASME OM Code, ISTC tests are performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The ASME OM Code, ISTC, provides criteria limits for the test parameters as identified in Table 3.9-13.

The detailed IST program includes the frequency and extent of disassembly and inspection based on suspected degradation of all safety-related check valves, including the basis for the frequency and the extent of each disassembly. Factors to be considered in the disassembly frequency and extent of disassembly include but are not limited to the following:

a. Performance history of the safety-related check valves to identify valves that are prone to degradation/wear

b. Analysis of trends of valve test parameters and service conditions

c. Analysis of valve components that are subject to aging and require an approach to maintenance and replacement
d. Results of non-intrusive valve testing. Use of non-intrusive technologies may obviate the need for inspection/disassembly of safety-related check valves altogether, provided the technologies demonstrate an equivalent ability to detect check valve degradation as inspection/disassembly would.

The program may be revised throughout plant life to minimize disassembly based on past disassembly experience.

3.9.6.3.4 Pressure Isolation Valve Leak Testing

The leak-tight integrity is verified for each valve relied on to provide a leak-tight function. These valves include:

a. Pressure-isolation valves (PIVs) that provide isolation of a pressure differential from one part of a system to another part or between systems. The leak testing of PIVs associated with the RCS is defined in NUREG-1482, Rev. 2 (Reference 58). The RCS PIVs are listed in Table 3.9-14 and are tested in accordance with Table 3.9-13 and Technical Specifications surveillance requirement 3.4.13.1.

b. Temperature-isolation valves (TIVs) whose leakage may cause unacceptable thermal stress, fatigue, or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps. Safety-related valves performing this duty are listed in Table 3.9-13, along with a description of specific leakage test requirements.

3.9.6.3.5 Containment Isolation Valve Leak Testing

The leak-tight integrity is verified for each valve relied on to provide a leak-tight function. These valves include containment-isolation valves (CIVs) that provide isolation capability for the piping systems penetrating containment.

CIVs are leak tested in accordance with 10 CFR Part 50, Appendix J. CIVs are listed along with their required testing in Table 6.2.4-1. Those CIVs for which a Type-C leakage rate test is specified in Table 6.2.4-1 are also tested in accordance with ASME OM Code, ISTC. These CIVs are designated in Table 3.9-13 by the valve function CIC. Those CIVs for which a Type-C leakage rate test is not specified in Table 6.2.4-1 are designated in Table 3.9-13 by the valve function CIN. The CIN valve function designation indicates
that these valves are listed in Table 6.2.4-1, but are not leakage rate tested in Tables 6.2.4-1 and 3.9-13.

3.9.6.3.6 Inservice Testing Program for Safety and Relief Valves

Pressure-relief devices are tested in accordance with ASME OM Code, including Appendix I to the OM Code, for IST.

Stroke tests are performed for dual-function safety and relief valves. Power-operated relief valves subject to the IST program are tested in accordance with Subsection ISTC-5100 for Category B valves and Subsection ISTC-5240 for Category C valves. The test equipment, including gauges, transducers, load cells, and calibration standards, used to determine valve set pressure is acceptable if the overall combined accuracy does not exceed ±1 percent of the indicated (measured) set pressure.

A list of safety and relief valves included in the IST program is provided in Table 3.9-13.

3.9.6.3.7 Inservice Testing Program for Manually Operated Valves

Safety-related active manually operated valves are identified in the IST Program Plan, and exercised periodically in accordance with frequency and requirements specified in the ASME OM Code.

Manual valves are exercised at least every 2 years. Exercise of a manual valve includes a complete cycle from open to fully closed.

A list of manual valves included in the IST program is provided in Table 3.9-13.

3.9.6.3.8 Inservice Testing Program for Explosively Activated Valves

Explosively activated valves are not included in the APR1400 design.

3.9.6.4 Inservice Testing Program for Dynamic Restraints

Safety-related systems inside and outside of containment may experience dynamic effects under various accident conditions, including seismic events and DBAs. Snubbers are attached to these systems to reduce these dynamic effects in areas where rigid supports are unacceptable. The snubber is selected to satisfy the system design requirements. The
snubber design and operating information form the basis for snubber examination and testing requirements.

As described in Subsection 3.12.6.6, dynamic restraints within piping systems is to be minimized as much as possible due to the maintenance and testing requirements for these components. However, dynamic restraints in the form of snubber supports are used where free thermal movements are required and restraining movements caused by dynamic loadings are also required. Snubber operability inspections and tests including scope and frequency requirements are specified and controlled in the components support inspection and testing program plan. The ASME OM Code, provides ISI methods and requirements for examinations and tests of snubbers at nuclear power plants.

Preservice and in-service examinations are performed using the VT-3 visual examination method described in IWA-2213 of the ASME Section XI, 2007 edition with 2008 addenda. Snubbers are visually examined to identify impaired function caused by physical damage, leakage, corrosion, or degradation from environmental exposure or operating conditions. External features that may affect operability are also examined. Intervals with the low rate of problem are in accordance with ASME Code Case OMN-13 (Reference 67) as accepted in NRC RG 1.192 (Reference 64).

Preservice functional testing is performed on snubbers prior to initial plant operation. This testing may be performed at the manufacturer’s facility. Inservice functional testing is performed over the test plan intervals specified in ASME OM Code, ISTD. Snubbers are tested in their installed location or removed and bench tested. Snubbers are tested in their as-found condition and the test parameters are selected so that the snubbers are tested to the fullest extent practicable.

The functional test for snubbers is performed to verify activation level of velocity or acceleration, release rate, and drag force.

Generic Letter 90-09 (Reference 68) addresses that a snubber is considered unacceptable if it fails the acceptance criteria of the visual inspection. An engineering evaluation will be conducted to determine the cause of unacceptability for an unacceptable snubber. The unacceptable snubbers will be adjusted, repaired, modified, or replaced.

The APR1400 snubber design incorporates accessibility provisions for maintenance, inspection, and testing of components. The correct installation and operation of snubbers is confirmed as part of the ITP described in Section 14.2. This program includes visual
inspections, hot and cold position measurements, and documenting thermally induced component movement that occurs during plant startup.

The methods addressed in Nonmandatory Appendix F of the ASME OM Code (Reference 33) will be applied for service life monitoring of dynamic restraints.

The COL applicant is to provide a table of all safety-related components that use snubbers in support systems and that includes the following information (COL 3.9(6)):

a. Identification of the systems and components that use snubbers

b. The number of snubbers used in each system and on the components in that system

c. Identification of the type(s) of snubber (hydraulic or mechanical)

d. Specification whether the snubber was constructed in accordance with the ASME Section III, NF

e. A statement of whether the snubber is used as a shock, vibration, or dual-purpose snubber

f. If a snubber is identified as a dual-purpose or vibration arrester type, indication of whether the snubber and/or component was evaluated for fatigue strength

3.9.6.5 Relief Requests and Alternative Authorizations to ASME OM Code

In case implementing the full requirements of the ASME OM Code is impractical, the relief requests from the testing requirements of the ASME OM Code will be made on a case-by-case basis by the COL Applicant. The following information should be described in the relief requests.

(1) Identification of the component by name, functions, ASME Section III Code Class, and valve category as defined in ISTC of the ASME OM Code and pump group as defined in ISTB of the ASME OM Code.

(2) Identification of the ASME OM Code requirements from which the applicant is requesting relief.
(3) For a relief request under 10 CFR 50.55a(f)(6) or (g)(6), the basis for requesting the relief and an explanation of why compliance with the ASME OM Code is impractical.

Any alternative request will be also submitted by the COL applicant as applicable. For an alternative request pursuant to which 10 CFR 50.55a(z), details regarding the proposed alternative(s) demonstrating that (1) the proposed IST will provide an acceptable level of quality and safety, or (2) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

ASME OM Code Cases will be implemented as part of the IST program for pumps, valves, and dynamic restraints as accepted in NRC RG 1.192, if applicable.

3.9.7 [Reserved]

3.9.8 [Reserved]

3.9.9 Combined License Information

COL 3.9(1) The COL applicant is to provide the inspection results for the APR1400 reactor internals classified as non-prototype Category I in accordance with NRC RG 1.20.

COL 3.9(2) The COL applicant is to identify the site-specific active pumps.

COL 3.9(3) The COL applicant is to provide a full description of the IST program (including PST and MOV testing) for pumps, valves, and dynamic restraints that will be administratively controlled such that the applicable requirements of the ASME OM Code edition and addenda are incorporated in the IST program.

COL 3.9(4) The COL applicant is to provide an IST program including the type of testing and frequency of site-specific pumps subject to IST in accordance with the ASME OM Code and Table 3.9-13.

COL 3.9(5) The COL applicant is to provide an IST program including the type of testing and frequency of any site-specific valves subject to IST in accordance with the ASME OM Code and Table 3.9-13.
COL 3.9(6) The COL applicant is to provide a table listing all safety-related components that use snubbers in their support systems.

COL 3.9(7) The COL applicant is to review the detailed analysis of mechanical tanks, including the effects of fluid sloshing.

3.9.10 References


20. RELAP5/MOD3.1, Transient Hydraulic Analysis Program, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho, USA.


   Loss-of-Coolant Accident Conditions with Application of Analysis to C-E 800 MWe


27. CENPD-252-P-A, “Method for the Analysis of Blowdown Induced Forces in a Reactor


   Blowdown Analysis (Modifications),” Combustion Engineering, Inc., Supplement 2,
   February 1975.

30. Scherer, A. E., Licensing Manager, (C-E), Letter to D. F. Ross, Assistant Director of
    Reactor Safety Division of Systems Safety, LD-76-026, March 1976 (Proprietary).

31. Parr, O. D., Chief Light Water Reactor Project Branch 1-3, Division of Reactor
    Licensing (NRC), Letter to F. M. Stern, Vice President of Projects (C-E), June 1975.

32. Kniel, K., Chief Light Water Reactors Branch No. 2, Letter to A. E. Scherer, Licensing
    Manager (C-E), August 1976 (Staff Evaluation of CENPD-213).

33. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants,”
    American Society of Mechanical Engineers, the 2004 Edition with the 2005 and 2006
    Addenda.

34. Regulatory Guide 1.100, “Seismic Qualification of Electric and Mechanical
    Equipment for Nuclear Power Plants,” Rev. 3, U.S. Nuclear Regulatory Commission,
    September 2009.


### Transients Used in Stress Analysis

<table>
<thead>
<tr>
<th>Event Conditions</th>
<th>Event Description</th>
<th>Specific Events</th>
<th>Occurrences for Design Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Level A Service Conditions</strong></td>
<td></td>
<td></td>
<td>60 Years(^{(1)})</td>
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<tr>
<td>Normal Event-1A</td>
<td>Steady-state operation with normal parameter variations in the increasing direction (5 to 100 %)</td>
<td>3,200</td>
<td>1,000,000</td>
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<tr>
<td>Normal Event-1B</td>
<td>Steady-state operation with normal parameter variations in the decreasing direction (5 to 100 %)</td>
<td>3,200</td>
<td>1,000,000</td>
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<tr>
<td>Normal Event-2A</td>
<td>Daily load follow operation (from 100 to 50 % power)(^{(2)})</td>
<td>22,000</td>
<td>15,000</td>
</tr>
<tr>
<td>Normal Event-2B</td>
<td>Daily load follow operation (from 50 to 100 % power)(^{(2)})</td>
<td>22,000</td>
<td>15,000</td>
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<td>Normal Event-3A</td>
<td>Turbine step load change in the increasing direction</td>
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<td>Turbine power steps of +10 percent (15 to 100 % power)</td>
<td>3,200</td>
<td>2,000</td>
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<td></td>
<td>Turbine power steps of +1 percent (5 to 15 % power)</td>
<td>1,600</td>
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<td></td>
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<td>Normal Event-3B</td>
<td>Turbine step load change in the decreasing direction</td>
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<td>Turbine power steps of -10 percent (15 to 100 % power)</td>
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<td>Turbine power steps of -1 percent (5 to 15 % power)</td>
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<td>Normal Event-3C</td>
<td>Large turbine load step decrease</td>
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<td>Turbine load rejection up to 50 % (50 to 100 % power)</td>
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<td>Turbine generator runback to house load</td>
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<tr>
<td></td>
<td>Reactor trip</td>
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<td>Turbine trip</td>
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### Table 3.9-1 (2 of 7)

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<th>Event Conditions</th>
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<td>60 Years$^{(1)}$</td>
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<td>Normal Event-4A</td>
<td>Turbine ramp load change in the increasing direction</td>
<td>Turbine power ramps of +5 %/min (15 to 100 % power)</td>
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<td>Turbine power ramps of +1 %/min (5 to 15 % power)</td>
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<td>(Total)</td>
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<td>Normal Event-4B</td>
<td>Turbine ramp load change in the decreasing direction</td>
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<td>Turbine power ramps of -1 %/min (5 to 15 % power)</td>
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<td>Loss of a main feedwater pump without reactor trip</td>
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<td>(Total)</td>
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<td>Normal Event-5</td>
<td>Non-load change events (planned)</td>
<td>NSSS operations with the control systems in the manual mode of the CEA, turbine bypass valves, pressurizer spray/heaters, pressurizer level control and feedwater flow (0 to 5 % power)</td>
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<td>Opening or closure of the feedwater economizer valve</td>
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<td>NSSS operations with the control systems in the manual mode (5 to 100 % power)</td>
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<td>Manual operation of the auxiliary spray system</td>
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<td>High steam generator blowdown</td>
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<td>Shift from normal to maximum CVCS flow rate and return</td>
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<td></td>
<td></td>
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<td>60 Years(1)</td>
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<td>40 Years(1)</td>
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<td>Normal Event-6</td>
<td>Non-load change events</td>
<td>Low-low VCT level and charging pump diversion to the boric acid storage tank</td>
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<td></td>
<td>(unplanned)</td>
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<tr>
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<td>Spurious actuation of the pressurizer spray</td>
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<tr>
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<td>Spurious actuation of the pressurizer heaters</td>
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<td></td>
<td></td>
<td>Inadvertent closure of one economizer or downcomer feedwater valve</td>
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<tr>
<td></td>
<td></td>
<td>Inadvertent opening of one economizer or downcomer feedwater valve</td>
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<td></td>
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<tr>
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<td></td>
<td>Inadvertent isolation of one main feedwater heater</td>
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<td></td>
<td></td>
<td>240</td>
</tr>
<tr>
<td>Normal Event-7</td>
<td>Plant events below power</td>
<td>Startup and coastdown of a reactor coolant pump at HSB</td>
<td>2,000</td>
</tr>
<tr>
<td></td>
<td>operation</td>
<td></td>
<td>1,340</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Startup and shutdown of the shutdown cooling system at HSD</td>
<td>250</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>170</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Spurious startup of a safety injection pump during shutdown conditions</td>
<td>60</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>40</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Spurious actuation of the pressurizer heaters at HSB</td>
<td>60</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>40</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Total)</td>
<td>2,370</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>1,590</td>
</tr>
<tr>
<td>Normal Event-8</td>
<td>Plant heatup</td>
<td></td>
<td>250</td>
</tr>
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<td></td>
<td></td>
<td></td>
<td>170</td>
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<tr>
<td>Normal Event-9</td>
<td>Plant cooldown</td>
<td></td>
<td>250</td>
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<td></td>
<td></td>
<td></td>
<td>170</td>
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## Table 3.9-1 (4 of 7)

<table>
<thead>
<tr>
<th>Event Conditions</th>
<th>Event Description</th>
<th>Specific Events</th>
<th>Occurrences for Design Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>60 Years (1)</td>
</tr>
<tr>
<td>Level B Service Conditions</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Upset Event-1</strong></td>
<td>Increase heat removal by the secondary system</td>
<td>Decrease in feedwater temperature</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Increase in feedwater flow rate</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Increase in steam flow rate</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Inadvertent opening of a main steam safety valve</td>
<td>10</td>
</tr>
<tr>
<td></td>
<td></td>
<td><strong>(Total)</strong></td>
<td>70</td>
</tr>
<tr>
<td><strong>Upset Event-2</strong></td>
<td>Decrease heat removal by the secondary system</td>
<td>Loss of external load</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Loss of condenser vacuum</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Loss of non-emergency AC power to the station auxiliaries</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Main steam isolation valve closure</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Loss of normal feedwater flow</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td><strong>(Total)</strong></td>
<td>100</td>
</tr>
<tr>
<td><strong>Upset Event-3</strong></td>
<td>Decrease in reactor coolant system flow rate</td>
<td>Loss of forced reactor coolant flow</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Natural circulation cooldown (HSB to HSD)</td>
<td>10</td>
</tr>
<tr>
<td></td>
<td></td>
<td><strong>(Total)</strong></td>
<td>30</td>
</tr>
</tbody>
</table>
## Table 3.9-1 (5 of 7)

<table>
<thead>
<tr>
<th>Event Conditions</th>
<th>Event Description</th>
<th>Specific Events</th>
<th>Occurrences for Design Purpose</th>
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</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>60 Years$^{(1)}$</td>
</tr>
<tr>
<td>Upset Event-4</td>
<td>Reactivity and power distribution anomalies</td>
<td>Uncontrolled CEA withdrawal at low power</td>
<td>5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Uncontrolled CEA withdrawal at high power</td>
<td>5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Control rod misoperation, RPCS inadvertent operation, or operator error</td>
<td>50</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Total)</td>
<td>60</td>
</tr>
<tr>
<td>Upset Event-5</td>
<td>Increase in reactor coolant system inventory</td>
<td>Loss of component cooling water to the letdown heat exchanger</td>
<td>10</td>
</tr>
<tr>
<td></td>
<td></td>
<td>CVCS malfunction that increases RCS inventory</td>
<td>10</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Total)</td>
<td>20</td>
</tr>
<tr>
<td>Upset Event-6</td>
<td>Decrease in reactor coolant system inventory</td>
<td>Inadvertent opening of a pilot operated safety and relief valve (POSRV closed as expected)</td>
<td>10</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Failure of small lines carrying coolant outside containment (letdown line break)</td>
<td>20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Total)</td>
<td>30</td>
</tr>
<tr>
<td>Upset Event-7</td>
<td>Reactor coolant pump seal failure</td>
<td></td>
<td>10</td>
</tr>
<tr>
<td>Upset Event-8</td>
<td>Loss of seal injection with loss of cooling water</td>
<td></td>
<td>5</td>
</tr>
</tbody>
</table>
### Event Conditions

**Level C Service Conditions**

<table>
<thead>
<tr>
<th>Event</th>
<th>Description</th>
<th>Specific Events</th>
<th>Occurrences for Design Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>none</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Level D Service Conditions**

<table>
<thead>
<tr>
<th>Event</th>
<th>Description</th>
<th>Specific Events</th>
<th>Occurrences for Design Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td>Faulted Event-1</td>
<td>Increase heat removal by the secondary system</td>
<td>Steam system piping failure</td>
<td>1</td>
</tr>
<tr>
<td>Faulted Event-2</td>
<td>Decrease heat removal by the secondary system</td>
<td>Feedwater system pipe break (FWLB)</td>
<td>1</td>
</tr>
<tr>
<td>Faulted Event-3</td>
<td>Decrease in reactor coolant system flow rate</td>
<td>Reactor coolant pump rotor seizure</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reactor coolant pump shaft break</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Total)</td>
<td>2</td>
</tr>
<tr>
<td>Faulted Event-4</td>
<td>Reactivity and power distribution anomalies</td>
<td>Rod ejection accident</td>
<td>1</td>
</tr>
<tr>
<td>Faulted Event-5</td>
<td>Decrease in reactor coolant system inventory</td>
<td>Inadvertent opening of a pilot operated safety and relief valve (POSRV fails to close)</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Steam generator tube rupture (SGTR)</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Loss of coolant accidents resulting from postulated pipe breaks within the RCS pressure boundary (LOCA)</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Total)</td>
<td>3</td>
</tr>
<tr>
<td>Faulted Event-6</td>
<td>Total loss of feedwater flow</td>
<td></td>
<td>1</td>
</tr>
</tbody>
</table>
## APR1400 DCD TIER 2

Table 3.9-1 (7 of 7)

<table>
<thead>
<tr>
<th>Event Conditions</th>
<th>Event Description</th>
<th>Specific Events</th>
<th>Occurrences for Design Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>60 Years&lt;sup&gt;(1)&lt;/sup&gt;</td>
</tr>
<tr>
<td>Test Conditions</td>
<td></td>
<td></td>
<td>15</td>
</tr>
<tr>
<td></td>
<td>RCS hydrostatic test</td>
<td></td>
<td>15</td>
</tr>
<tr>
<td></td>
<td>Secondary hydrostatic test</td>
<td></td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>RCS leak test</td>
<td></td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>Secondary leak test</td>
<td></td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>SIS/SCS preoperational and maintenance test</td>
<td></td>
<td>360</td>
</tr>
<tr>
<td></td>
<td>SIS/SCS check valve operability test</td>
<td></td>
<td>120</td>
</tr>
</tbody>
</table>

(1) The design life for RCS main components and Class 1 piping is 60 years, and the design life for Class 2 and 3 piping and other components except RCS main components is 40 years.

(2) Although APR1400 will be operated as a base load plant, the effects of daily load follow operation are accounted for in the structural design and analysis of ASME Code Class 1 components, reactor internals, and component supports.
Table 3.9-2

**Loading Combinations for ASME Code Class 1, 2, and 3 Components**

<table>
<thead>
<tr>
<th>Condition</th>
<th>Design Loading (2) Combination</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design</td>
<td>PD + DW + IRWST</td>
</tr>
<tr>
<td>Level A (Normal) (3)</td>
<td>PO + DW</td>
</tr>
<tr>
<td>Level B (Upset) (3)</td>
<td>PO + DW + IRWST + DFL</td>
</tr>
<tr>
<td>Level C (Emergency)</td>
<td>No Loads (5)</td>
</tr>
<tr>
<td>Level D (Faulted)</td>
<td>PO + DW + SRSS(SSE + DF + IRWST) (4)</td>
</tr>
</tbody>
</table>

(1) For piping, see Tables 3.9-10, 3.12-1, and 3.12-2.

(2) Legend:
- PD = design pressure
- PO = operating pressure
- DW = deadweight
- SSE = safe shutdown earthquake
- DF = dynamic system loadings associated with pipe breaks (not eliminated by a leak-before-break analysis), or POSRV actuation
- IRWST = In-containment refueling water storage tank discharge loads
- DFL = Dynamic fluid loads are occasional loads associated with hydraulic transients caused by events such as valve actuation (safety or relief valve discharge, rapid valve opening/closing), water hammer, or steam hammer.

(3) As required by the ASME Section III, other loads, such as thermal transient, and thermal gradient, require consideration in addition to the primary stress producing loads listed. SSE is considered in equipment fatigue evaluations in accordance with Subsection 3.7.3.1.

(4) Detailed loading combinations of ASME Code Class 1 components and component supports are described in Subsection 3.9.3.1.

(5) Refer to Subsection 3.9.3.1.

(6) Internal operating and/or design pressure loading is applicable to pressure boundary components and not to their supports.
Table 3.9-3

Stress Limits for ASME Code Class 1 Components, Piping, and Component Supports

<table>
<thead>
<tr>
<th></th>
<th>Component and Piping Stress Limits(1)</th>
<th>Component Support Stress Limits(2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design</td>
<td>NB-3221, NB-3231, and NB-3652</td>
<td>Table NF-3131(a) - 1</td>
</tr>
<tr>
<td>Level A (Normal)</td>
<td>NB-3222, NB-3232, and NB-3653</td>
<td>Table NF-3131(a) - 1</td>
</tr>
<tr>
<td>Level B (Upset)</td>
<td>NB-3223, NB-3233, and NB-3654</td>
<td>Table NF-3131(a) - 1</td>
</tr>
<tr>
<td>Level C (Emergency)</td>
<td>NB-3224, NB-3234, and NB-3655</td>
<td>Table NF-3131(a) - 1</td>
</tr>
<tr>
<td>Level D (Faulted)(3)</td>
<td>NB-3225, NB-3235, and NB-3656</td>
<td>Table NF-3131(a) - 1</td>
</tr>
</tbody>
</table>

(1) Stress limits listed are used as required by the ASME Section III, and applicable addenda for all components except active components. Active components are designed to the stress limits of NB-3221 and NB-3231 for design conditions and the stress limits of NB-3222 and NB-3232 for all other conditions for active components.

(2) Stress limits used are as required by the ASME Section III and modified by NRC RGs 1.124 and 1.130.

(3) For faulted condition loadings, bolts in the load path connecting two members of an NF support for Class 1 components are designed in accordance with Appendix F of the ASME Section III for friction type connections with tensile stresses limited to the lesser of 0.7 $S_u$ or $S_y$. 
# Seismic Category I Active Valves

<table>
<thead>
<tr>
<th>Valve No.</th>
<th>System Name (Safety Function)</th>
<th>Valve Type</th>
<th>ASME Section III Class</th>
<th>Actuator Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>SI-100</td>
<td>IRWST recirculation isolation (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-101</td>
<td>IRWST recirculation isolation (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-113</td>
<td>SIS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-123</td>
<td>SIS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-133</td>
<td>SIS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-143</td>
<td>SIS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-157</td>
<td>CSS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-158</td>
<td>CSS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-159</td>
<td>SC pump suction check</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-160</td>
<td>SC pump suction check</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-168</td>
<td>SCS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-178</td>
<td>SCS (operate)</td>
<td>Check</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-179</td>
<td>SCS (operate)</td>
<td>Relief</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-189</td>
<td>SCS (operate)</td>
<td>Relief</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>SI-215</td>
<td>SI tank (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-217</td>
<td>SI system (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-225</td>
<td>SI tank (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-227</td>
<td>SI system (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-235</td>
<td>SI tank (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-237</td>
<td>SI system (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-245</td>
<td>SI tank (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-247</td>
<td>SI system (operate)</td>
<td>Check</td>
<td>1</td>
<td>None</td>
</tr>
<tr>
<td>SI-300</td>
<td>CS/SCS IRWST recirculation isolation</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-301</td>
<td>CS/SCS IRWST recirculation isolation</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-302</td>
<td>SI IRWST recirculation isolation</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-303</td>
<td>SI IRWST recirculation isolation</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-304</td>
<td>IRWST isolation</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-305</td>
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<td>Gate</td>
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<td>Motor</td>
</tr>
<tr>
<td>SI-308</td>
<td>IRWST isolation</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-309</td>
<td>IRWST isolation</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-310</td>
<td>SCS 1 flow control (operate)</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
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</table>
Table 3.9-4 (2 of 22)

<table>
<thead>
<tr>
<th>Valve No.</th>
<th>System Name (Safety Function) (1)(2)(3)</th>
<th>Valve Type</th>
<th>ASME Section III Class</th>
<th>Actuator Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>SI-311</td>
<td>SCS 2 flow control (operate)</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-312</td>
<td>SDCHX bypass (operate)</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-313</td>
<td>SDCHX bypass (operate)</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-314</td>
<td>SCS 1 IRWST recirculation line flow control</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-315</td>
<td>SCS 2 IRWST recirculation line flow control</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-321</td>
<td>Hot leg injection (operate)</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-322</td>
<td>Hot leg injection leakage return (close)</td>
<td>Globe</td>
<td>1</td>
<td>Pneumatic</td>
</tr>
<tr>
<td>SI-331</td>
<td>Hot leg injection (operate)</td>
<td>Globe</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-332</td>
<td>Hot leg injection leakage return (close)</td>
<td>Globe</td>
<td>1</td>
<td>Pneumatic</td>
</tr>
<tr>
<td>SI-340</td>
<td>SCS/CSS pump suction cross connection (close)</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-341</td>
<td>SCS/CSS pump discharge cross connection (operate)</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-342</td>
<td>SCS/CSS pump suction cross connection (close)</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-343</td>
<td>SCS/CSS pump discharge cross connection (close)</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-344</td>
<td>SC pump suction isolation (close)</td>
<td>Gate</td>
<td>2</td>
<td>Motor</td>
</tr>
<tr>
<td>SI-346</td>
<td>SC pump suction isolation (close)</td>
<td>Gate</td>
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Table 3.9-4 (9 of 22)

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Table 3.9-4 (15 of 22)

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<td>SX-1004</td>
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Table 3.9-4 (22 of 22)

<table>
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<th>Valve No.</th>
<th>System Name (Safety Function)</th>
<th>Valve Type</th>
<th>ASME Section III Class</th>
<th>Actuator Type</th>
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(1) “Operate” is defined as valve being capable of both opening and closing.
(2) “Close” is defined as valve being capable of moving to or maintaining a closed position.
(3) “Open” is defined as valve being capable of moving to or maintaining an open position.
Table 3.9-5

**Stress Criteria for Safety-Related**

**ASME Section III, Class 2 and Class 3 Vessels**

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Stress Limits&lt;sup&gt;(1)&lt;/sup&gt;</th>
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<tbody>
<tr>
<td>Design and Level A</td>
<td>$\sigma_m \leq 1.0 \ S $</td>
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<tr>
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<td>$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5 \ S$</td>
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<tr>
<td>Level B</td>
<td>$\sigma_m \leq 1.1 \ S $</td>
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<tr>
<td></td>
<td>$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \ S$</td>
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<tr>
<td>Level C</td>
<td>$\sigma_m \leq 1.5 \ S $</td>
</tr>
<tr>
<td></td>
<td>$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80 \ S$</td>
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<tr>
<td>Level D</td>
<td>$\sigma_m \leq 2.0 \ S $</td>
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<tr>
<td></td>
<td>$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4 \ S$</td>
</tr>
</tbody>
</table>

<sup>(1)</sup> Stress limits are taken from ASME Section III, NC/ND, Table 3321-1.
### Table 3.9-6

**Stress Criteria for ASME Section III, Class 2 and Class 3 Non-active Pumps**

<table>
<thead>
<tr>
<th>Plant Condition</th>
<th>Service Limits(^{(1)})</th>
<th>Loads</th>
<th>Stress Limits(^{(2)})</th>
<th>(P_{\text{max}}) (^{(3)})</th>
<th>Subsections(^{(5)})</th>
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<tr>
<td>Design</td>
<td>Design</td>
<td>Sustained loads: pressure, weight, other mechanical loads</td>
<td>(\sigma_m \leq 1.0) S ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5) S</td>
<td>-</td>
<td>ASME Section III, NC/ND-3400</td>
</tr>
<tr>
<td>Normal</td>
<td>Level A</td>
<td>Sustained loads: pressure, weight, other mechanical loads</td>
<td>(\sigma_m \leq 1.0) S ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5) S</td>
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<td>Upset</td>
<td>Level B</td>
<td>Occupational loads: pressure, weight, thermal effects, dynamic fluid loads(^{(4)})</td>
<td>(\sigma_m \leq 1.1) S ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65) S</td>
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<td>Emergency</td>
<td>Level C</td>
<td>No loads (Refer to Subsection 3.9.3.1)</td>
<td>(\sigma_m \leq 1.5) S ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.8) S</td>
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<td>Faulted</td>
<td>Level D</td>
<td>Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,(^{(4)}) SSE inertia, pipe break</td>
<td>(\sigma_m \leq 2.0) S ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4) S</td>
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<td>ASME Section III, NC/ND-3400</td>
</tr>
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</table>

\(^{(1)}\) Service limits are taken from ASME Section III, NCA-2142.4.

\(^{(2)}\) Stress limits are taken from ASME Section III, NC and ND, Table NC/ND-3416-1.

\(^{(3)}\) The maximum pressure does not exceed the tabulated factors listed under \(P_{\text{max}}\) times the design pressure.

\(^{(4)}\) Dynamic fluid loads (DFL) are occasional loads such as safety and relief valve thrust, steam hammer, water hammer, or other loads associated with plant upset or faulted condition as applicable. Dynamic loads are combined by the SRSS method.

# APR1400 DCD TIER 2

## Table 3.9-7

Stress Criteria for ASME, Section III Class 2 and Class 3 Active Pumps

<table>
<thead>
<tr>
<th>Plant Condition</th>
<th>Service Limits(^{(1)})</th>
<th>Loads</th>
<th>Stress Limits(^{(2)})</th>
<th>(P_{\text{max}}) (^{(3)})</th>
<th>Subsections(^{(5)})</th>
</tr>
</thead>
</table>
| Design          | Design                  | Sustained loads: pressure, weight, other mechanical loads | \(\sigma_m \leq 1.0 \text{ S} \)  
\( (\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5 \text{ S} \) | - | ASME Section III, NC/ND-3400 |
| Normal          | Level A                 | Sustained loads: pressure, weight, other mechanical loads | \(\sigma_m \leq 1.0 \text{ S} \)  
\( (\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5 \text{ S} \) | 1.0 | ASME Section III, NC/ND-3400 |
| Upset           | Level B                 | Occupational loads: pressure, weight, thermal effects, dynamic fluid loads\(^{(4)}\) | \(\sigma_m \leq 1.1 \text{ S} \)  
\( (\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \text{ S} \) | 1.1 | ASME Section III, NC/ND-3400 |
| Emergency       | Level B                 | No loads (Refer to Subsection 3.9.3.1) | \(\sigma_m \leq 1.1 \text{ S} \)  
\( (\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \text{ S} \) | 1.1 | ASME Section III, NC/ND-3400 |
| Faulted         | Level B                 | Occupational loads: pressure, weight, thermal effects, dynamic fluid loads\(^{(4)}\), SSE inertia, pipe break | \(\sigma_m \leq 1.1 \text{ S} \)  
\( (\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \text{ S} \) | 1.1 | ASME Section III, NC/ND-3400 |

\(^{(1)}\) Service limits are taken from ASME Section III, NCA-2142.4.

\(^{(2)}\) Stress limits are taken from ASME Section III, NC and ND, Table NC/ND-3416-1. However, the stress limits for service level C and D are more restrictive than the ASME Section III limits to provide reasonable assurance of pump operability.

\(^{(3)}\) The maximum pressure does not exceed the tabulated factors listed under \(P_{\text{max}}\) times the design pressure.

\(^{(4)}\) Dynamic fluid loads (DFLs) are occasional loads such as safety and relief valve thrust, steam hammer, water hammer, or other loads associated with plant upset or faulted condition as applicable. Dynamic loads are combined by the SRSS method.

### Stress Criteria for Safety-Related ASME Section III, Class 2 and Class 3 Non-active Valves

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Stress Limits (^{(1)-(4)-(6)})</th>
<th>(P_{\text{max}} (^{(5)}))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design and Level A</td>
<td>(\sigma_m \leq 1.0 \ S)</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>((\sigma_m \ or \ \sigma_L) + \sigma_b \leq 1.5 \ S)</td>
<td></td>
</tr>
<tr>
<td>Level B</td>
<td>(\sigma_m \leq 1.1 \ S)</td>
<td>1.1</td>
</tr>
<tr>
<td></td>
<td>((\sigma_m \ or \ \sigma_L) + \sigma_b \leq 1.65 \ S)</td>
<td></td>
</tr>
<tr>
<td>Level C</td>
<td>(\sigma_m \leq 1.5 \ S)</td>
<td>1.2</td>
</tr>
<tr>
<td></td>
<td>((\sigma_m \ or \ \sigma_L) + \sigma_b \leq 1.8 \ S)</td>
<td></td>
</tr>
<tr>
<td>Level D</td>
<td>(\sigma_m \leq 2.0 \ S)</td>
<td>1.5</td>
</tr>
<tr>
<td></td>
<td>((\sigma_m \ or \ \sigma_L) + \sigma_b \leq 2.4 \ S)</td>
<td></td>
</tr>
</tbody>
</table>

(1) Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied:

a) The section modulus and areas of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles.

b) Code allowable stress, \(S\), for valve body material is equal to or greater than the code allowable stress, \(S\), of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above is multiplied by the ratio of \(S_{\text{pipe}}/S_{\text{valve}}\). If unable to conform with this requirement, the design by analysis procedure of the ASME Section III, NB-3545.2, is an acceptable alternate method.

(2) Casting quality factor of 1.0 is used.

(3) These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

(4) Design requirements listed in this table are not applicable to valve stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet.

(5) The maximum pressure resulting from Service Levels B, C, or D does not exceed the tabulated factors listed under \(P_{\text{max}}\) times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

(6) Stress limits are taken from ASME Section III, Table NC/ND-3521-1.
### Stress Criteria for ASME Section III, Class 2 and Class 3 Active Valves

<table>
<thead>
<tr>
<th>Service Loading Conditions (Plant Condition)</th>
<th>Service Limits(^{(1)})</th>
<th>Stress Limits(^{(2)})</th>
<th>(P_{\text{max}}) (^{(3)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design</td>
<td>Design</td>
<td>(\sigma_m \leq 1.0 \ S) ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5 \ S)</td>
<td>-</td>
</tr>
<tr>
<td>Level A (normal)</td>
<td>Level A</td>
<td>(\sigma_m \leq 1.0 \ S) ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5 \ S)</td>
<td>1.0</td>
</tr>
<tr>
<td>Level B (upset)</td>
<td>Level B</td>
<td>(\sigma_m \leq 1.1 \ S) ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \ S)</td>
<td>1.1</td>
</tr>
<tr>
<td>Level C (emergency)</td>
<td>Level B</td>
<td>(\sigma_m \leq 1.1 \ S) ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \ S)</td>
<td>1.1</td>
</tr>
<tr>
<td>Level D (faulted)</td>
<td>Level B</td>
<td>(\sigma_m \leq 1.1 \ S) ((\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 \ S)</td>
<td>1.1</td>
</tr>
</tbody>
</table>

\(^{(1)}\) Service Limits (Level A, B, C, and D) are defined in ASME Section III, NCA-2142.4.

\(^{(2)}\) Stress limits are in accordance with ASME Section III, Table NC/ND-3521-1. For service loading conditions level C and level D, the stress limits specified for active valves are more restrictive than ASME Section III limits to assure the operability of valves.

\(^{(3)}\) The maximum pressure does not exceed the tabulated factors listed under \(P_{\text{max}}\) times the design pressure or the rated pressure at the applicable operating condition temperature.

\(^{(4)}\) Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied:

a) The section modulus and areas of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles.

b) Code allowable stress, \(S\), for valve body material is equal to or greater than the code allowable stress, \(S\), of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above is multiplied by the ratio of \(S_{\text{pipe}}/S_{\text{valve}}\). If unable to conform with this requirement, the design by analysis procedure of ASME Section III, NB-3545.2, is an acceptable alternate method.

\(^{(5)}\) Casting quality factor of 1.0 is used.

\(^{(6)}\) Design requirements listed in this table are not applicable to valve stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet.
### Table 3.9-10

**Loading Conditions and Load Combinations Requirements for ASME Section III, Class 1, 2, and 3 Piping Supports**

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Loading Combination</th>
</tr>
</thead>
<tbody>
<tr>
<td>Level A</td>
<td>Weight, Thermal(^{(1)}), Friction(^{(5)})</td>
</tr>
<tr>
<td>Level B</td>
<td>Weight, Thermal(^{(1)}), IRWST discharge loads (^{(3)}), Dynamic fluid loads(^{(2)}), Friction(^{(5)})</td>
</tr>
<tr>
<td>Level C</td>
<td>No loads (Refer to Subsection 3.9.3.1)</td>
</tr>
<tr>
<td>Level D</td>
<td>Weight, Thermal(^{(1)}), IRWST discharge loads (^{(3)}), Dynamic fluid loads(^{(2)}), SSE inertia, SSE seismic movements, Pipe break loads(^{(4)}), Friction(^{(5)})</td>
</tr>
</tbody>
</table>

1. Thermal conditions (including ambient temperature) to be combined to provide maximum load combinations.
2. Dynamic fluid loads due to safety/relief valve thrust, steam hammer, and water hammer.
3. In-containment refueling water storage tank discharge loads.
4. Pipe break loads include loads due to LOCA.
5. Friction forces included are from the movement of piping due to thermal expansion with sum of deadweight and applicable signed loads.
Table 3.9-11

Stress Limits for CEDM Pressure Housings

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Stress Categories and Limits of Stress Intensities(^{(1)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Design: design pressure, normal operating loads(^{(3)}), IRWST discharge loads</td>
<td>NB-3221 and Figure NB-3221-1 including notes</td>
</tr>
<tr>
<td>2. Level A: normal operating loads, normal operating transients</td>
<td>NB-3222 and Figure NB-3222-1 including notes</td>
</tr>
<tr>
<td>3. Level B: normal operating loads, upset transients, IRWST discharge loads, fatigue loads due to SSE(^{(3)})</td>
<td>NB-3223 and Figure NB-3222-1 including notes</td>
</tr>
<tr>
<td>4. Level C: no loads (Refer to Subsection 3.9.3.1)</td>
<td>NB-3224 and Figure NB-3224-1 including notes</td>
</tr>
<tr>
<td>5. Level D(^{(4)}): operating pressure, normal operating loads, IRWST discharge loads, BLPB loads, SSE loads</td>
<td>Appendix F Article F-1000 Rules for evaluation of service conditions loading with level D service limits</td>
</tr>
<tr>
<td>6. Testing: testing plant transients</td>
<td>NB-3226</td>
</tr>
</tbody>
</table>

\(^{(1)}\) References listed are taken from ASME Section III.

\(^{(2)}\) “Normal operating loads” is defined in Subsection 3.9.4.3.

\(^{(3)}\) Fatigue loads due to SSE are applied in accordance with Subsection 3.9.2.2.3.

\(^{(4)}\) SSE loads is combined with BLPB loads and IRWST discharge loads by the SRSS method in accordance with the guidelines of NUREG-0484.
Table 3.9-12

Stress Limits for Reactor Internals Design and Service Loads

<table>
<thead>
<tr>
<th>Stress Limit</th>
<th>Description</th>
</tr>
</thead>
</table>
| Design Limits           | The reactor internals are designed to meet the design limits defined in ASME Section III, NG-3221, for design loadings. The reactor internals are safety Class 3 and seismic Category I in accordance with ANSI/ANS 51.1.  
Core support structures are constructed in accordance with ASME Section III, NG-1100. The reactor internals other than core support structures meet the guidelines of ASME Section III, NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures.  
Under Level D service loadings, the maximum stress intensity is obtained from principal stresses resulting from an SRSS combination of IRWST, BLPB, and SSE plus normal operating dynamic and static loading in accordance with NUREG-0484, Rev. 1. For other than Level D service loading conditions, maximum stress intensity are derived from an SRSS combination of dynamic loads in accordance with NUREG-0484, Rev. 1, or a more conservative summation of stress intensities. |
| Level A Service Limits  | The reactor internals are designed to meet the Level A service limits defined in ASME Section III, NG-3222, for Level A service loadings.                                                                               |
| Level B Service Limits  | The reactor internals are designed to meet the Level B service limits defined in ASME Section III, NG-3223, for Level B service loadings.                                                                           |
| Level C Service Limits  | There are no Level C service loadings. Refer to Subsection 3.9.3.1.                                                                                                                                         |
| Level D Service Limits  | The reactor internals are designed to meet the Level D service limits defined in ASME Section III, NG-3225, for elastic system analysis of Appendix F of ASME Section III using Level D service loadings. Maximum stress intensity is obtained from principal stresses resulting from an SRSS combination of IRWST, BLPB, and SSE loadings plus normal operation loads in accordance with NUREG-0484, Rev. 1. |
### Table 3.9-13 (1 of 85)

#### Inservice Testing of Safety-Related Pumps and Valves

<table>
<thead>
<tr>
<th>Pump No.</th>
<th>Pump Description</th>
<th>Pump Type</th>
<th>Safety Class</th>
<th>OM Code Group</th>
<th>Test Parameter (b)</th>
<th>Test Freq</th>
<th>Test Criteria</th>
<th>Figure No.</th>
</tr>
</thead>
<tbody>
<tr>
<td>CC-PP01A</td>
<td>Component Cooling Water pump 1A</td>
<td>Centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.2-1</td>
</tr>
<tr>
<td>CC-PP01B</td>
<td>Component Cooling Water pump 1B</td>
<td>Centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.2-1</td>
</tr>
<tr>
<td>CC-PP02A</td>
<td>Component Cooling Water pump 2A</td>
<td>Centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.2-1</td>
</tr>
<tr>
<td>CC-PP02B</td>
<td>Component Cooling Water pump 2B</td>
<td>Centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.2-1</td>
</tr>
<tr>
<td>CC-PP03A</td>
<td>Component Cooling Water makeup pump 3A</td>
<td>Centrifugal</td>
<td>3</td>
<td>B</td>
<td>DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.2-1</td>
</tr>
<tr>
<td>CC-PP03B</td>
<td>Component Cooling Water makeup pump 3B</td>
<td>Centrifugal</td>
<td>3</td>
<td>B</td>
<td>DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.2-1</td>
</tr>
<tr>
<td>SI-PP02A</td>
<td>Safety Injection pump 1</td>
<td>Centrifugal</td>
<td>2</td>
<td>B</td>
<td>DP, Q, V (40)</td>
<td>3 mo</td>
<td>Table ISTB-5100-1 in ASME OM Code</td>
<td>6.3.2-1</td>
</tr>
<tr>
<td>SI-PP02B</td>
<td>Safety Injection pump 2</td>
<td>Centrifugal</td>
<td>2</td>
<td>B</td>
<td>DP, Q, V (40)</td>
<td>3 mo</td>
<td>Table ISTB-5100-1 in ASME OM Code</td>
<td>6.3.2-1</td>
</tr>
<tr>
<td>SI-PP02C</td>
<td>Safety Injection pump 3</td>
<td>Centrifugal</td>
<td>2</td>
<td>B</td>
<td>DP, Q, V (40)</td>
<td>3 mo</td>
<td>Table ISTB-5100-1 in ASME OM Code</td>
<td>6.3.2-1</td>
</tr>
<tr>
<td>SI-PP02D</td>
<td>Safety Injection pump 4</td>
<td>Centrifugal</td>
<td>2</td>
<td>B</td>
<td>DP, Q, V (40)</td>
<td>3 mo</td>
<td>Table ISTB-5100-1 in ASME OM Code</td>
<td>6.3.2-1</td>
</tr>
<tr>
<td>SI-PP01A</td>
<td>Shutdown Cooling pump 1</td>
<td>Centrifugal vertical</td>
<td>2</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5100-1 in ASME OM Code</td>
<td>6.3.2-1</td>
</tr>
<tr>
<td>SI-PP01A</td>
<td>Shutdown Cooling pump 2</td>
<td>Centrifugal vertical</td>
<td>2</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5100-1 in ASME OM Code</td>
<td>6.3.2-1</td>
</tr>
<tr>
<td>CS-PP01A</td>
<td>Containment Spray pump 1</td>
<td>Centrifugal</td>
<td>2</td>
<td>B</td>
<td>DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>6.2.2-1</td>
</tr>
<tr>
<td>CS-PP01B</td>
<td>Containment Spray pump 2</td>
<td>Centrifugal</td>
<td>2</td>
<td>B</td>
<td>DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>6.2.2-1</td>
</tr>
<tr>
<td>SX-PP01A</td>
<td>Essential Service Water pump 1A</td>
<td>Vertical line shaft centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.1-1</td>
</tr>
<tr>
<td>SX-PP01B</td>
<td>Essential Service Water pump 1B</td>
<td>Vertical line shaft centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.1-1</td>
</tr>
<tr>
<td>SX-PP02A</td>
<td>Essential Service Water pump 2A</td>
<td>Vertical line shaft centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.1-1</td>
</tr>
<tr>
<td>SX-PP02B</td>
<td>Essential Service Water pump 2B</td>
<td>Vertical line shaft centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.1-1</td>
</tr>
<tr>
<td>FC-PP01A</td>
<td>Spent Fuel Pool cooling pump A</td>
<td>Centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.1.3-1</td>
</tr>
<tr>
<td>FC-PP01B</td>
<td>Spent Fuel Pool cooling pump B</td>
<td>Centrifugal</td>
<td>3</td>
<td>A</td>
<td>DP, Q, V</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.1.3-1</td>
</tr>
<tr>
<td>AF-PP02A</td>
<td>Motor Driven AFW pump PP02A</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>B</td>
<td>N, DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>10.4.9-1</td>
</tr>
<tr>
<td>Pump No</td>
<td>Pump Description</td>
<td>Pump Type</td>
<td>Safety Class</td>
<td>OM Code Group</td>
<td>Test Parameter (1)</td>
<td>Test Freq</td>
<td>Acceptance Criteria</td>
<td>Figure No.</td>
</tr>
<tr>
<td>---------</td>
<td>-------------------------------------------------------</td>
<td>----------------------</td>
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</tr>
<tr>
<td>AF-PP01A</td>
<td>TD AFW pump PP01A</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>B</td>
<td>N, DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>10.4.9-1</td>
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<tr>
<td>AF-PP02B</td>
<td>MD AFW pump PP02B</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>B</td>
<td>N, DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>10.4.9-1</td>
</tr>
<tr>
<td>AF-PP01B</td>
<td>TD AFW pump PP01B</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>B</td>
<td>N, DP, Q</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>10.4.9-1</td>
</tr>
<tr>
<td>WO-PP01A</td>
<td>ECW pump PP01A</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>A</td>
<td>N, DP, Q, V, P₀</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.7-1</td>
</tr>
<tr>
<td>WO-PP02A</td>
<td>ECW pump PP02A</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>A</td>
<td>N, DP, Q, V, P₀</td>
<td>3 mo</td>
<td>Table ISTB-5121-1 in ASME OM Code</td>
<td>9.2.7-1</td>
</tr>
<tr>
<td>WO-PP03A</td>
<td>ECW makeup pump PP03A</td>
<td>Centrifugal horizontal</td>
<td>3</td>
<td>B</td>
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<td>3 mo</td>
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### APR1400 DCD TIER 2

Table 3.9-13 (6 of 85)

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3.9-158  
Rev. 3
Table 3.9-13 (8 of 85)

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Rev. 3
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### APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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Table 3.9-13 (13 of 85)
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3.9-169

Rev. 3
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**APR1400 DCD TIER 2**

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# APR1400 DCD TIER 2

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3.9-172 Rev. 3
### APR1400 DCD TIER 2

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3.9-173

Rev. 3
### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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- **Valve Type (a):** GT, GL
- **Valve Act (b):** EL, S
- **Safety Class (c):** 2, B
- **Code Cat (d):** A
- **Valve Funct (e):** CIC, S
- **Safety Position:** MT S
- **Test Reqd (f):** RO
- **Test Freq (g):** 3 mo, 2 yr
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### APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

### Table 3.9-13 (37 of 85)

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## APR1400 DCD TIER 2

### Table 3.9-13 (39 of 85)

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Rev. 3
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### Table 3.9-13 (45 of 85)

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3.9-201

Rev. 3
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# APR1400 DCD TIER 2

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## APR1400 DCD TIER 2

Table 3.9-13 (62 of 85)

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3.9-213

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<td>Valve No.</td>
<td>Valve Description</td>
<td>Valve Type</td>
<td>Valve Act</td>
<td>Safety Class</td>
<td>Code Cat</td>
<td>Valve Funct</td>
<td>Safety Position</td>
<td>Test Req'd</td>
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<td>Closed</td>
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<td>Open/Closed</td>
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<td>Open/Closed</td>
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<td>2 yr</td>
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## Table 3.9-13 (66 of 85)

<table>
<thead>
<tr>
<th>Valve No.</th>
<th>Valve Description</th>
<th>Valve Type</th>
<th>Valve Act</th>
<th>Safety Class</th>
<th>Code Cat</th>
<th>Valve Funct</th>
<th>Safety Position</th>
<th>Test Reqd</th>
<th>Test Freq</th>
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<td>GL</td>
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<td>3 mo 3 mo 2 yr 3 mo 2 yr</td>
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<td>CM-1013</td>
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<td>CK</td>
<td>SA</td>
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<td>A/C</td>
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<td>S RF LT</td>
<td>RO RO 2 yr</td>
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<td>CM-1014</td>
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<td>CK</td>
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<td>A/C</td>
<td>CIC</td>
<td>Closed</td>
<td>S RF LT</td>
<td>RO RO 2 yr</td>
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<tr>
<td>PS-0031</td>
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<td>GT</td>
<td>AD</td>
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<td>B</td>
<td>CIN</td>
<td>Closed</td>
<td>S MT FS LPV</td>
<td>3 mo 3 mo 2 yr 3 mo 2 yr</td>
</tr>
<tr>
<td>PS-0032</td>
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<td>S MT FS LPV</td>
<td>3 mo 3 mo 2 yr 3 mo 2 yr</td>
</tr>
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## APR1400 DCD TIER 2

### Table 3.9-13 (67 of 85)

<table>
<thead>
<tr>
<th>Valve No.</th>
<th>Valve Description</th>
<th>Valve Type (a)</th>
<th>Valve Act (b)</th>
<th>Safety Class (c)</th>
<th>Code Cat (d)</th>
<th>Valve Funct (e)</th>
<th>Safety Position</th>
<th>Test Reqd (f)</th>
<th>Test Freq (g)</th>
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<td>PS-0033</td>
<td>SG 1 downcomer sample isolation</td>
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<td>B</td>
<td>CIN</td>
<td>Closed</td>
<td>S</td>
<td>3 mo, 3 mo, 2 yr</td>
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<td>PS-0034</td>
<td>SG 2 downcomer sample isolation</td>
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<td>2</td>
<td>B</td>
<td>CIN</td>
<td>Closed</td>
<td>S</td>
<td>3 mo, 3 mo, 2 yr</td>
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<td>SG 1 blowdown cold leg sample isolation</td>
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<td>AD</td>
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<td>B</td>
<td>CIN</td>
<td>Closed</td>
<td>S</td>
<td>3 mo, 3 mo, 2 yr</td>
</tr>
<tr>
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<td>CIN</td>
<td>Closed</td>
<td>S</td>
<td>3 mo, 3 mo, 2 yr</td>
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<tr>
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<td>AD</td>
<td>2</td>
<td>B</td>
<td>CIN</td>
<td>Closed</td>
<td>S</td>
<td>3 mo, 3 mo, 2 yr</td>
</tr>
<tr>
<td>PS-0258</td>
<td>SG 2 blowdown hot leg sample isolation</td>
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<td>B</td>
<td>CIN</td>
<td>Closed</td>
<td>S</td>
<td>3 mo, 3 mo, 2 yr</td>
</tr>
</tbody>
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**Notes:**

(a) Valve Type:  
- GL - Globe  
- BF - Butterfly  
- GT - Gate  
- PK - Packless  
- CK - Check  
- PL - Plug  
- RV - Relief  
- POS - Pilot operated safety relief valve  
- 3W - 3Way  
- BL - Ball

(b) Valve Actuator:  
- EL - Electric motor  
- S - Solenoid  
- SA - Self actuating  
- EH - Electro-hydraulic  
- AD - Air diaphragm  
- P - Air operated with Piston  
- M - Manual

(c) Safety Classification as defined in Subsection 3.2.3.

(d) Valve ASME Code Category A, B, C, or D as defined in ASME OM Code, ISTC-1300.

(e) Valve Function:

<table>
<thead>
<tr>
<th>Code</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>CIC</td>
<td>Containment isolation valve as listed in Table 6.2.4-1, which is Type-C leakage rate tested in accordance with ANSI/ANS 56.8</td>
</tr>
<tr>
<td>CIN</td>
<td>Containment isolation valve as listed in Table 6.2.4-1, which is not Type-C leakage rate tested in accordance with ANSI/ANS 56.8</td>
</tr>
<tr>
<td>PIV</td>
<td>Pressure isolation valve</td>
</tr>
<tr>
<td>TIV</td>
<td>Temperature isolation valve</td>
</tr>
<tr>
<td>P</td>
<td>Passive valves as defined by ASME OM Code, ISTA-2000 are denoted by a P in this column. All other valves are active valves.</td>
</tr>
</tbody>
</table>
Table 3.9-13 (69 of 85)

(f) Required valve tests per ASME OM Code, ISTC and Mandatory Appendix I; and additional required testing:

- **LT** - Valve leakage rate test (per ASME OM Code, ISTC): Subsections ASME OM Code, ISTC for valves with function CIC in (e) above. Subsection ASME OM Code, ISTC for valves with function PIV in (e) above. Reactor coolant system PIVs are leakage rate tested in accordance with Technical Specifications Surveillance Requirement 3.4.13.1. Subsection ASME OM Code, ISTC for Category A valves except the valves with function TIV. Subsection ASME OM Code, ISTC for valves with function TIV.

- **LPV** - Valve position verification (ASME OM Code, ISTC)

- **S** - Valve stroke exercise in the forward flow direction:
  - Category A or B (ASME OM Code, ISTC)
  - Category C (ASME OM Code, ISTC)

- **RF** - Reverse flow exercise for A/C and C valves (ASME OM Code, ISTC). “RF” testing is performed at the same testing frequency as the corresponding "S" test, unless otherwise described.

- **MT** - Valve stroke time test of Category A or B power-operated valves (ASME OM Code, ISTC)

- **FS** - Valves test for fail-safe actuation of Category A or B valves (ASME OM Code, ISTC)

- **RVT** - Relief valve test (ASME OM Code, ISTC)
(g) Pump or valve test exclusions, alternatives, and frequency per ASME OM Code, ISTB and ASME OM Code, ISTC. For valves whose test frequency exceeds the normal frequency, see the note (as indicated in parenthesis beside the test frequency) for additional information/justification.

**CS** - Cold Shutdown
The following condition applies for all testing performed during cold shutdown:
Cold shutdown testing in accordance with the requirements of ASME OM Code, ISTC. See the note for additional information/justification. Valve exercising during cold shutdown commences until all testing is complete or the plant is ready to return to power. A completion of all valve testing is not a prerequisite to return to power. Any testing not completed by the end of one cold shutdown is performed during subsequent cold shutdowns, starting from the last test performed at the previous cold shutdown. In case of frequent shutdowns, testing is not performed more often than once every 3 months.

**RO** - Refueling Outage.
All refueling outage valve testing is to be completed prior to returning the plant to operation.

**RR** - Partially stroke valve at or when proceeding to/starting up from cold shutdown. Fully stroke valve during each refueling outage. Some valves may require mechanical exercising or disassembly during each refueling outage to verify operability. All RR testing measures are completed prior to returning the plant to operation.

**QC** - Partially stroke valve every 3 months. Fully stroke valve during cold shutdown.

**EI** - Valve operates in the course of plant operation at a frequency that satisfies test requirements. Additional exercising not required provided the test parameters are analyzed and recorded at an operational interval not exceeding the test interval requirement.
Category A or B (ASME OM Code, ISTC)
Category C (ASME OM Code, ISTC)
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(h) Pump test parameters as defined in ASME OM Code, ISTB-5000:

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<tr>
<th>Symbol</th>
<th>Parameter</th>
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<tbody>
<tr>
<td>N</td>
<td>Speed</td>
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<tr>
<td>DP</td>
<td>Differential Pressure</td>
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<tr>
<td>Q</td>
<td>Flow Rate</td>
</tr>
<tr>
<td>SPs</td>
<td>Static Suction Pressure</td>
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<tr>
<td>SPo</td>
<td>Operating Suction Pressure</td>
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<tr>
<td>SPc</td>
<td>Calculated Suction Pressure</td>
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</table>

Note: If ASME OM Code ISTB pump tests cannot be performed on the CCW or ESW pumps due to inability to repeat pump tests single-point flow conditions, pump curve testing will be used to assess pump degradation in accordance with ASME OM Code Case OMN-16, accepted for use in NRC RG 1.192(Rev.1), as described in Subsection 3.9.6.2.

(i) (Intentionally blank)

(1) Valves: CC-0231, CC-0249, CC-0250
During normal operations, these valves are open to supply/return cooling water to/from the reactor coolant pump (RCP) coolers. Failure of these valves in the closed position could lead to pump damage or failure and force a unit shutdown. Therefore, these valves will be tested during cold shutdown when the RCPs are not operating.

(2) Valves: CC-1099, CC-1100
These valves provide containment isolation and overpressure protection for the component cooling water (CCW) supply and return lines to/from the RCPs (refer to Figure 9.2.2-1). Since these CCW lines are to remain in service during plant operation, it is impractical to perform S or RF testing on the valves on a quarterly test frequency.

The reverse stroke (RF) test of CC-1099 is impractical to perform without isolating CC-1071 and CC-1070. Since this method of testing requires access to areas of high radiation and contamination, a test of this type can be performed only during refueling. This method of testing is the same as will be employed for the ANSI/ANS 56.8, Type-C leakage rate tests. Therefore, LT testing accomplishes and satisfies the reverse flow testing requirements for CC-1099.
The reverse stroke (RF) test of CC-1100 is impractical to perform without isolating CC-0249 and CC-1085. Since this method of testing requires access to areas of high radiation and contamination, a test of this type can be performed only during refueling. This method of testing is the same as will be employed for the ANSI/ANS 56.8, Type-C leakage rate tests. Therefore, LT testing accomplishes and satisfies the reverse flow testing requirements for CC-1100.

Valve CC-1100 is reverse flow stroke tested during Cold Shutdown. The forward stroke (S) of CC-1100, however, is impractical to perform without isolating CC-0249, CC-1084, and CC-1085. Since this valve testing methodology requires containment entries to areas of high radiation and contamination, the forward stroke testing of CC-1100 will be performed during refueling.

(3) Valves: CC-0143, CC-0144, CC-0145, CC-0146, CC-0147, CC-0148, CC-0149, CC-0150

These valves close on receipt of a safety injection actuation signal to isolate the non-essential component cooling water (CCW) loops. The non-essential cooling loops provide cooling of the non-essential chillers. When plant chilled water (PCW) is secured for testing of PCW valves and non-essential header CCW valves may then be stroke tested. These valves will use the same test frequency as the PCW valves, as described in (28).

The Division 1 non-essential CCW header services the letdown heat exchanger in addition to the Division 1 non-essential chillers. Closing the Division 1 non-essential CCW header valves during plant operation could result in unnecessary reactor coolant system transients. Also, failure to cool the high temperature letdown flow leaving the regenerative heat exchanger can lead to cavitation at the letdown orifices, which has been known to cause line failure. Therefore, valves CC-0143, CC-0145, CC-0147, and CC-0149 will be tested during cold shutdown.

(4) Valves: CC-0296, CC-0297, CC-0301, CC-0302

These valves isolate cooling water to/from the letdown heat exchanger and close on a containment isolation actuation signal. For reasons stated in Note (3) above, testing these valves during normal operations is not practical. Therefore, these valves will be tested during cold shutdown.
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(5) Valves: CC-1685, CC-1686

These valves provide containment isolation and overpressure protection for the component cooling water (CCW) supply and return lines to/from the letdown heat exchanger (refer to Figure 9.2.2-1 (4 of 4)). Since these CCW lines are to remain in service during plant operation, it is impractical to perform S or RF testing on the valves on a quarterly test frequency.

Valve CC-1685 is forward stroke tested during cold shutdown. The reverse stroke (RF) test of CC-1685, however, is impractical to perform without isolating CC-0297, CC-1683, and CC-1681. Since this method of testing requires access to areas of high radiation and contamination, a test of this type can be performed only during refueling. This method of testing is the same as will be employed for the ANSI/ANS 56.8, Type-C leakage rate tests. Therefore, LT testing accomplishes and satisfies the reverse flow testing requirement for CC-1685.

Valve CC-1686 is reverse flow stroke tested during cold shutdown. The forward stroke (S) of CC-1686, however, is impractical to perform without isolating CC-0301, CC-1682, and CC-1684. Since this valve testing methodology requires containment entries to areas of high radiation and contamination, the forward stroke testing of CC-1686 will be performed during refueling.

(6) Valves: CV-255

This valve isolates seal injection water to the RCP seals. Valve closure during normal operations with the RCPs operating would result in damage to pump seals. Therefore, this valve will be tested during cold shutdown when the RCPs are not operating.

(7) Valves: CV-505, CV-506

These valves close on receipt of a containment spray actuation signal to isolate the RCP seal return line. During normal operations, these valves are open to maintain seal injection flow across the RCP seals. Closure of these valves during normal operations would inhibit seal water flow across the RCP seals, which would result in damage to the pump seals. Therefore, these valves will be tested during cold shutdown when the RCPs are not operating.
Table 3.9-13 (74 of 85)

(8) Valves: CV-515, CV-516, CV-522, CV-523

These valves are normally open to pass letdown flow from the RCS to the chemical and volume control system (CVCS). Stroking these valves during normal operations could result in unnecessary RCS transients. In addition, these valves are subjected to high stresses when cycled due to the high-pressure environment in which they operate. Repeated cycling of the valves at this high pressure could severely affect valve integrity over the expected operating life of the valves. In addition, failure of these valves in the closed position could result in a loss of pressurizer level control, forcing a unit shutdown. Therefore, these valves will be tested during cold shutdown when the effects of valve operation are minimized. Globe valve CV-515 performs a temperature isolation valve (TIV) function. This valve isolates the letdown line on a high temperature, as sensed downstream of the letdown heat exchanger by dedicated temperature monitors (refer to Figure 9.3.4-1, Sheet 1).

The setpoints of these temperature monitors and associated valve isolation actuation circuitry are such that the design temperature limits of the interfacing CVCS piping and components will not be exceeded prior to the closure of CV-515. Temperature monitors are also used to evaluate the integrity of CV-515 in this closed position. Each refueling outage, an integrity evaluation of CV-515 is performed by isolating the letdown line using CV-515, and then subjecting the valve to reactor coolant system pressure and temperature and analyzing the resultant temperature differential across the valve over time. RCS pressure and temperature may be actually lower than plant at-power RCS pressure and temperature levels to avoid valve duty stress, provided these parameters are analyzed and extrapolated to full RCS pressure and temperature.

(9) Valves: CV-524

This valve functions as a containment isolation valve and isolates charging flow to the RCS. During normal operations, this charging flow is used to cool the letdown flow in the regenerative heat exchanger and to provide makeup to the RCS. In addition, failure of this valve in the closed position could result in a loss of pressurizer level control, forcing a unit shutdown. Therefore, this valve will be tested during refueling because it is not practical to test during normal operation and cold shutdown.

(10) Valves: CV-747

This valve functions as a containment isolation valve. Testing requires that charging flow be isolated. As stated in Note (9) above, this is not practical during normal operations. Therefore, this valve will be tested during refueling.

(11) Valves: CV-835

This valve functions as a containment isolation valve. Testing requires that seal injection to the RCPs be isolated. As stated in Note (6), this is not practical during normal operations. Therefore, this valve will be tested during cold shutdown.
(12) Valves: RG-0410, RG-0411, RG-0412, RG-0413, RG-0414, RG-0415, RG-0416, RG-0417

These valves are closed during normal plant operations to maintain the reactor coolant pressure boundary (RCPB). These valves are active valves and are designed to be used during a safety-grade cooldown of the RCS. Opening these valves during normal operation leaves only one Class 1 valve, which does not maintain the RCPB according to 10 CFR 50.2 and ANSI/ANS 51.1 definitions. While there is a third valve downstream of the two reactor coolant gas vent system (RCGVS) valves, the piping and the third valve are Class 2. In order to maintain the integrity of the RCPB, these valves are to be tested during plant shutdown periods only and not during reactor operation.

These valves will be tested as cold shutdown valves.


These valves isolate main feedwater to the SGs upon receipt of a main steam isolation signal (MSIS). Closure of these valves during normal operations would isolate feedwater to the SGs, which may result in a severe transient in the SG and a unit trip. Therefore, these valves will be tested during cold shutdown.

(14) Valves: MS-011, MS-012, MS-013, MS-014, MS-090, MS-091, MS-092, MS-093

These valves are main steam isolation valves (MSIVs) and main steam drip leg isolation valves, which isolate the main steam lines upon receipt of an MSIS. Performance of either a full-stroke or partial-stroke test during normal operations may cause severe transients in the main steam lines and result in a unit trip. The valves will therefore be full stroke tested on a cold shutdown frequency basis, but with the unit in Mode 3 and at operating temperature and pressure to replicate design conditions under which valve closure is to be achieved.

(15) Valves: SI-568, SI-569

These valves are to close to prevent reverse flow when either the SC/CS pumps are used to provide containment spray or to provide shutdown cooling flow. These valves are tested by operating the SC/CS pump in any train, opening the discharge crossover isolation valve between the two systems, and isolating the suction of the off-line pump. Closure of the check valve on reverse flow in the discharge of the off-line pump is verified by monitoring pressure increase upstream of the valve.

Check valves SI-113, SI-133, SI-404, SI-405, SI-434, SI-446, SI-540, and SI-542 are to be provided with sufficient flow from the SI pumps to stroke to their full-open position. The flow for stroke testing these valves passes through the DVI nozzles and into the RCS. The SI pump discharge pressure is not sufficient to overcome normal RCS operating pressure. In addition, any flow from safety injection through these valves and into the RCS during power operations would produce an undesirable temperature transient at the DVI nozzles. The valves are also not full or partial stroked during cold shutdown, since this may result in low-temperature overpressurization of the RCS. Since it is impractical to full stroke test these check valves during plant operation or to perform full/partial stroke test during cold shutdown conditions, these valves are full stroke tested each refueling outage.

Check valves SI-520, SI-522, SI-523, SI-532, and SI-533 are to be provided with sufficient flow from the SI pumps to stroke to their full-open position. The flow for stroke testing these valves passes through the RCS hot legs (hot leg injection). The SI pump discharge pressure is not sufficient to overcome normal RCS operating pressure. In addition, any flow from safety injection through these valves and into the RCS during power operations would produce an undesirable temperature transient at the shutdown cooling line connections to the hot legs. The valves are also not full or partial stroked during cold shutdown, since this may result in low-temperature overpressurization of the RCS. Since it is impractical to full stroke test these check valves during plant operation or to perform full/partial stroke test during cold shutdown conditions, these valves are full stroke tested each refueling outage.

Check valves SI-113 and SI-133 are not reverse flow tested quarterly, since testing of these valves during power operations would require containment entries by plant personnel to high-radiation and airborne contamination areas. These valves are not reverse flow tested every cold shutdown because of the extensive test equipment setup, which could extend the cold shutdown. These valves are reverse flow tested during refueling.

(16A) Valves: SI-404, SI-405, SI-434, SI-446

These valves are reverse flow tested by pressurizing the volume of piping between these valves and their respective SI pump discharge maintenance isolation valve (SI-476, SI-478, SI-435, and SI-447) with water, and using either pressure decay or volumetric analysis to determine valve reverse seating function.
Check valves SI-123, SI-143, SI-541, SI-543, SI-168, and SI-178 are to be provided with sufficient flow to stroke to their full-open position. This test flow ultimately passes through the DVI nozzles and into the RCS. Neither the SI nor the SC/CS pump discharge pressures are sufficient to overcome normal RCS operating pressure in order to establish the flow required to perform a partial or full stroke test of these valves. In addition, any flow from safety injection or shutdown cooling to the RCS during power operations would produce an undesirable temperature transient at the DVI nozzles. During cold shutdown, the SI pumps may not be used for stroke testing these valves, because this could result in low-temperature overpressurization of the reactor vessel. A full flow stroke test of these valves during cold shutdown is achievable by use of the SC/CS pumps. Valves SI-123 and SI-143 are not reverse flow tested quarterly, since testing of these valves during power operations would require containment entries by testing personnel to high radiation and airborne contamination areas. These valves are not reverse flow tested every cold shutdown because of the extensive test equipment setup which could extend the cold shutdown. These valves are reverse flow tested during refueling.

(18) Valves: CS-1007, CS-1008, CS-1014
These valves are required to open to pass flow from the containment spray (CS) pumps to the containment atmosphere. These valves cannot be stroked open with CS flow, since this would result in spraying down containment. These valves will be equipped with external means to exercise the valve obturator and to measure the force required to exercise the valve open and closed (this performs both the S and RF test). Since the valves are located in a containment area subject to moderate to high radiation and contamination levels, the valves will be exercised each refueling outage, instead of during cold shutdown or plant operation (every 3 months).

These SIT outlet check valves are to be provided with sufficient flow from the SITs to the RCS to stroke to their full-open position. During normal operations, the SI tanks are not capable of providing flow to the RCS, due to RCS pressure and tank pressure limitations. Also, providing flow to the DVI nozzles during plant operations would cause undesirable temperature transients at the DVI nozzles. The SITs may be used, however, to provide flow to partially stroke these valves, with minimal temperature transient impact to the DVI nozzles, when proceeding to or starting up from cold shutdown. In this configuration, a full-flow stroke test is impractical due to significant inventory additions to the RCS should the SIT water level become too low. During refueling and with the reactor head removed, full-flow testing of these valves is practical. In this condition, SIT flow may be obtained, which is sufficient to full stroke the SIT outlet check valve, with minimal risk of injecting nitrogen into the RCS. Therefore, these valves will be partially stroke tested during cold shutdown and full-stroke tested during refueling.
Providing flow to the DVI nozzles in order to stroke these check valves during plant operation is not practicable, since RCS pressure during normal operations is significantly higher than the discharge pressures of the SI pumps, SC/CS pumps, or SITs. In addition, any flow from these sources to the RCS during power operations would produce an undesirable temperature transient at the DVI nozzles.

Check Valves SI-217 and SI-237:
Full-stroke testing of these check valves is not practical at cold shutdown for several reasons. First, SI pumps 4 and 3 are not capable of providing sufficient flow to full stroke their respective DVI check valve (SI-217/SI-237). Secondly, such use of SI pumps during cold shutdown condition is not practical, since it could result in low-temperature overpressurization of the reactor vessel. Thirdly, use of water inventory from their respective SITs to full-stroke test these check valves during cold shutdown when RCS pressure is low is impractical, because of the risk of injecting nitrogen into the RCS, and because the RCS is not capable of accepting the added SIT inventory from a full-stroke test.

However, a partial-stroke test of these check valves may be achieved with minimal temperature transient impact to their respective DVI nozzles, when proceeding to or starting up from cold shutdown, by use of water inventory from their respective SITs to establish flow through these valves. During refueling and with the reactor head removed, full flow testing of these valves is practical. In this condition, SIT flow may be obtained that is sufficient to full stroke the respective DVI check valve, with minimal risk of injecting nitrogen into the RCS. Therefore, these valves will be partially stroke tested during cold shutdown and full-stroke tested during refueling.

Check Valves SI-227 and SI-247:
These check valves have the same testing limitations as SI-217 and SI-237, above, except that a full-stroke test of the check valves is practical during cold shutdown by operating their respective SC/CS pump. SI-227 and SI-247 are tested as cold shutdown valves.

(21) Valves: SI-302, SI-303
Closing of these valves to perform stroke testing renders both SI pumps in the respective valve trains inoperable, which is in violation of Technical Specifications 3.5.2 and 3.5.3. Technical Specification 3.5.3, however, allows inoperability of both train SI pumps during refueling subject to prescribed RCS parameters.

(22) Valves: SI-424, SI-426, SI-448, SI-451
Each of these valves is reverse flow tested by isolating its associated pump and operating the other divisional SI pump in miniflow to provide reverse flow against the tested valve. In this test alignment, the operating pump is to remain in miniflow condition, fully capable of supplying design basis accident flow. Procedural measures are implemented so that only one train SI pump will be inoperable (i.e., the isolated SI pump).
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(23) Valves: CV-362, CV-363

These valves function as containment isolation valves and isolate shutdown cooling purification line. The valves are in the closed position during normal operation and open during shutdown cooling purification. Therefore, these valves will be tested during cold shutdown. To provide reasonable assurance of the operability of CV-363 for its containment isolation function, this valve is reverse flow tested. Reverse flow testing of CV-363 is not practical quarterly, since during unit operation, opening of CV-362 or other venting path could result in an inter-system LOCA.

The appropriate interval for such testing is during cold shutdown when the shutdown purification line is secured prior to unit startup.


These valves are to be open to pass flow from the SITs to the RCS. Technical Specifications do not permit testing these valves during normal operation since all four SITs are to be operable. Normal shutdown/startup procedures require these valves to be closed when proceeding to cold shutdown and to be opened when starting up from cold shutdown. Testing of these valves will be performed at this time.


These valves are to open to align the SC/CS pump suction to the RCS. These valves are interlocked such that they cannot be opened when RCS pressure is above the operating pressure of the SCS. Therefore, these valves cannot be tested during normal operations. Testing will be performed during cold shutdown when valves can be manipulated.

(26) Valves: VQ-0011, VQ-0012, VQ-0013, VQ-0014

These valves are to close on receipt of a CIAS to perform their containment isolation function. During normal operations, these valves are closed and Technical Specifications do not permit opening. Therefore, these valves will be tested during cold shutdown.

(27) Valves: SD-1115, SD-1116

These valves are to close on reverse flow in the SG wet layup recirculation line to perform their containment isolation function. The recirculation lines are isolated during normal operations and are only used when the SGs are in wet layup conditions such as during cold shutdown. Therefore, these valves will be tested during cold shutdown when the recirculation system is stopped.
(28) Valves: WI-012, WI-013, WI-015

During normal operations, these valves are open to provide plant chilled water (PCW) to the containment. Stroke testing of any one of these valves will require interruption of at least one division of PCW to the containment. To maintain containment air temperatures within the 120 °F Technical Specifications limit year-round, the containment coolers are required to operate with the two units in standby. There may be periods during the year, however, when two out of four containment cooler operations (one PCW division operating, one PCW division secured) provides sufficient cooling to maintain containment temperature within the Technical Specifications limit, due to less severe site climate and heat sink characteristics (e.g., non-summer months). For these periods of the year, the valves will be stroke tested quarterly.

For other periods of the year during which at least two of the four containment coolers are to be kept in operation to maintain containment temperature within the 48.9 °C (120 °F) Technical Specification limit, the valves will be tested as cold shutdown valves.


These main feedwater system check valves are located on the feedwater inlet lines to the steam generators. These check valves have only a safety function to close. Since these valves are to remain open during power operations to maintain steam generator level and prevent reactor trip and plant shutdown, quarterly reverse flow testing is impractical. As described in Section 10.4.7 and Figure 10.4.7-1, the feedwater split between the economizer feedwater lines and the downcomer feedwater line always maintains some flow through the downcomer feedwater line, even though the two economizer feedwater lines are sized to collectively provide 100 percent required flow to the steam generator they service. The 10 percent of required steam generator flow that passes through the downcomer line at full-power operation is used to maintain the downcomer line at a constant temperature to protect the line from thermal transient (water hammer) damage. Thus, the downcomer feedwater line may not be isolated during power operations in order to perform a reverse flow test on its feedwater check valves. Flow testing of these valves is performed on a cold shutdown (CS) frequency basis while the plant is in Mode 3 (hot standby), at which condition adequate steam generator pressure exists to perform reverse flow tests on the valves.

(30) (Intentionally Blank)
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These solenoid valves are both stroke tested (S) and fail-safe tested (FS) during cold shutdown because opening any of these valves will result in depressurizing the affected SIT, thus causing the SIT to be inoperable. These valves cannot be tested during plant operations, since plant Technical Specification 3.5.1 requires all SITs to remain operable in Mode 1 (power operations). Technical Specification LCO 3.5.1 (required action) for inoperability of any SIT requires restoration of that SIT to operable status in one hour, or commence unit shutdown. Since this LCO is too stringent to allow valve stroke or fail-safe testing of these valves during plant operations, this testing will be performed during cold shutdown.

(32) Valves: FP-1440

The safety function of valves FP-1440 in the forward stroke direction is to relieve thermal pressure to the containment fire water supply piping and thus prevent damage to the containment penetration as a result of containment heatup following a LOCA. It is impractical to perform a forward stroke test for check valves FP-1440 during power operations or cold shutdown for several reasons: significant radiation and contamination exposure to test personnel in containment, the necessity of disabling the sprinkler system within containment to perform the test, which jeopardizes system response to containment fire, and the extensive restoration/draining of the fire supply headers inside of containment post-testing to their normal “dry” status, which would result in extending the cold shutdown.

The reverse flow safety function is containment isolation. Reverse flow testing of these check valves is impractical during power operations or cold shutdown for several reasons: significant radiation and contamination exposure to test personnel in containment, the necessity of disabling the sprinkler systems to fill the “dry” fire water supply piping in the reverse flow test volume in order to establish backpressure on the check valve seat, which jeopardizes system response to containment fire, and the extensive restoration/draining of the fire supply headers inside of containment post-testing to their normal “dry” status, which would result in extending the cold shutdown.

(33) Valves: WM-1752

The safety function of valve WM-1752 in the forward stroke direction is to relieve thermal pressure to the containment demineralized water piping as a result of containment heatup following a LOCA. Verification of this safety function requires forward stroke testing, and use of demineralized water within containment. However, during power operations and cold shutdown, there are no users of demineralized water within containment to establish this flow, without necessitating containment entry to areas of high radiation dose and airborne contamination present during power operations and cold shutdown to manipulate manual valves at decontamination sinks, etc. The forward stroke test will be performed during refueling for ALARA purposes. Similarly, the RF test will require containment entry to areas of high radiation dose and airborne contamination present during power operations and cold shutdown. For ALARA purposes, this test will be performed during refueling.
(34) Valves: IW-0001, IW-0002, IW-0003, IW-0004

Valves IW-0001 and IW-0002 are motor-operated holdup volume tank (HVT) flooding valves; valves IW-0003 and IW-0004 are reactor cavity flooding valves. The valves are normally closed and remain closed throughout the recovery period of any design basis accident. The valves are opened either individually or simultaneously only for a severe accident, which requires flooding of the reactor cavity in the event of the reactor vessel breach. The opening of the valves allows water to flow from the IRWST to the reactor cavity to cover core debris. Operability of the valves is not required for shutting down the reactor, maintaining cold shutdown, or mitigating the consequences of any design basis accident.

Testing of the HVT flooding valves requires that the manual valves located upstream be closed to prevent the flow of water from the IRWST to the HVT. Closing the manual valves is not practical during operations at power because containment entry would be required. The reactor cavity flooding valves are not tested during operations at power because inadvertent actuation of the valves would result in the simultaneous opening of the reactor cavity flooding valves and the HVT flooding valves. Even though the IRWST water volume is designed to prevent breaching the reactor vessel, abnormal condition which results in flooding the HVT and the reactor cavity should be prevented. The HVT flooding valves and the reactor cavity flooding valves will be tested during each refueling outage. This will limit personnel radiation exposure and prevent the simultaneous opening of the reactor cavity flooding valves and the HVT flooding valves.


These air-operated SIT nitrogen pressure control valves are stroked in the course of plant operation as a matter of normal operation and pressure control of the SITs at a frequency that satisfies test requirements of quarterly testing. Fail-safe (FS) actuation on a 3-month basis, however, is impractical during plant operations (quarterly test frequency) or cold shutdown because such testing involves entries to containment to proximity of the SITs (high radiation dose and airborne contamination area) to fail air to the air diaphragm valve actuators. Therefore, the FS test for these valves will be performed on a refueling outage basis for ALARA purposes.


These air-operated valves are stroked on a quarterly frequency. Fail-safe (FS) actuation testing on a 3-month basis; however, is impractical during plant operations (quarterly test frequency) or cold shutdown because such testing involves entries to containment to proximity of the SITs (high radiation dose and airborne contamination area) to fail air to the air diaphragm valve actuators. Therefore, the FS test for these valves will be performed on a refueling outage basis for ALARA purposes.

(37) Although these emergency diesel generator support system components are Safety Class 3, they are procured, tested, and maintained as part of the emergency diesel generators themselves, which are tested for operability and reliability by the plant Technical Specifications. Therefore, these components are tested by Technical Specifications Surveillance Requirements of Technical Specification Section 3.8.
(38) Pressure isolation valves (PIVs) are not reverse flow tested quarterly, since testing of these valves during power operation would require containment entries to high radiation and airborne contamination areas. PIVs are not reverse flow tested every cold shutdown, because of the extensive test equipment setup, which could extend the cold shutdown. The RF function is verified, however, by leakage testing each valve in the reverse flow direction during unit startup for the testing frequency outlined in Technical Specification Surveillance Requirement 3.4.13.1. This surveillance requirement states that leakage testing of these valves is required every 18 months and prior to entering Mode 2 whenever the plant has been in Mode 5 (cold shutdown) for 7 days or more, if leakage testing has not been performed in the previous 9 months and within 24 hours following valve actuation due to automatic or manual action or flow through the valve(s).

(39) Inservice Testing/Monitoring for Valves on Piping Connected to the Steam Generator Secondary Side

Steam generator blowdown valves are tested for gross leakage each refueling outage. Testing is performed by isolating these valves individually against steam generator pressure and then monitoring the steam generator blowdown tank for an increase in tank level, which would be indicative of gross valve leakage.

Steam generator sampling line valves are tested for gross leakage each refueling outage. Testing is performed by isolating these valves individually against steam generator pressure and then monitoring sample line flow for gross valve leakage.

There is potential of thermal stratification in Auxiliary Feedwater (AF) piping, if the Feedwater (FW) back leakage occurs through the inside-containment containment isolation check valves. Therefore, the AF containment isolation valves are tested for gross leakage each refueling outage. The AF isolation check valves are leakage tested by individually subjecting these valves to steam generator pressures experienced during unit startup/shutdown and then measuring resultant valve leakage through the provided test connection. The outside-containment AF isolation valves are leakage tested by pressurizing the piping between these valves and their inside-containment containment isolation check valves while the steam generators are at startup/shutdown pressures. Valve leakage is then measured through the provided test connection. These AF valves also employ installed temperature instrumentation to detect leakage past these valves.
(40) Safety Injection System

For inservice testing of the safety injection pumps during refueling outages, a walkdown visual examination of safety injection system piping and components outside containment will be conducted to verify the leak-tight integrity of the system.

(41) Valves: CV-576, CV-577

These valves limit charging flow to RCS. In addition, failure of these valves in the closed position could result in a loss of pressurizer level control. Therefore, these valves will be tested during refueling because it is not practical to test during normal operation and cold shutdown.

(42) Valve: PX-1005

This valve is installed to isolate the containment building and protect the overpressure of sample collecting piping of the post-accident primary sampling system (Figure 9.3.2-1). PX-1005 is closed for normal plant operation, and the right-direction and reverse-direction stroke test of PX-1005 is operated by using test-fitting. The right-direction and reverse-direction stroke test of PX-1005 is not operated every quarter because an operator has to enter the high-radiation and radiation contamination air-particle areas in the containment building during power operations. This valve is not tested during cold shutdown because cold shutdown can be extended by the installation of test equipment but tested during refueling operation.

(43) Valve: PX-1020

This valve is installed to isolate the containment building of sample collecting piping of containment atmosphere (Figure 9.3.2-1). PX-1020 is closed for normal plant operation, and the right-direction and reverse-direction stroke test of PX-1020 is operated by using test-fitting. The right-direction and reverse-direction stroke test of PX-1020 is not operated every quarter because an operator has to enter the high-radiation and radiation contamination air-particle areas in the containment building during power operations. This valve is not tested during cold shutdown because cold shutdown can be extended by the installation of test equipment but tested during refueling operation.

When auxiliary feedwater (AF) system operation is required, these valves are to open to provide flow to the steam generators (SGs).
Testing of these valves requires AF injection into the SGs, which is not practical during normal operations due to the effects of thermal shock to the SG feedwater nozzles and potential overcooling of the RCS. Testing during cold shutdown is not desirable because the SG is in wet layup conditions. Therefore, these valves will be tested following cold shutdown prior to entering Mode 2, which allows normal SG water levels to be established and the system aligned for standby readiness.
Reverse flow for reverse flow testing of valves AF-1008A, AF-1008B, AF-1007A, and AF-1007B is obtained by opening AF pump to AF isolation valves AF-0043, AF-0044, AF-0045, and AF-0046, respectively.

(45) (Intentionally Blank)

Valve: RC-0385, RC-0386
If POSRVs open with POSRV discharge path to the S/G compartments for testing of RC-0385 and RC-0386, reactor coolant will be discharged to the S/G compartments and the containment will be under radioactive contamination environment. This condition limits containment entries by plant personnel when the valves or their indication fail during the testing of RC-0385 and RC-0386. Therefore, testing will be performed during refueling for ALARA purposes.

Valve: FC-1145
Testing of the check valve FC-1145 is not performed quarterly since testing of this valve during power operations would require containment entries by plant personnel to high-radiation and airborne contamination areas. Therefore, forward and reverse stroke tests of this valve are performed during refueling.
Table 3.9-14

RCS Pressure Isolation Valves Associated with the Reactor Coolant System

Detailed testing information is contained in Table 3.9-13 and Technical Specification Surveillance Requirement 3.4.13.1.

<table>
<thead>
<tr>
<th>Valve</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>SI-215</td>
<td>SI tank 4 discharge check</td>
</tr>
<tr>
<td>SI-217</td>
<td>DVI nozzle 1B check valve</td>
</tr>
<tr>
<td>SI-225</td>
<td>SI tank 2 discharge check</td>
</tr>
<tr>
<td>SI-227</td>
<td>DVI nozzle 2B check valve</td>
</tr>
<tr>
<td>SI-235</td>
<td>SI tank 3 discharge check</td>
</tr>
<tr>
<td>SI-237</td>
<td>DVI nozzle 2A check valve</td>
</tr>
<tr>
<td>SI-245</td>
<td>SI tank 1 discharge check</td>
</tr>
<tr>
<td>SI-247</td>
<td>DVI nozzle 1A check valve</td>
</tr>
<tr>
<td>SI-322</td>
<td>Hot leg injection 1 bleed-off isolation</td>
</tr>
<tr>
<td>SI-332</td>
<td>Hot leg injection 2 bleed-off isolation</td>
</tr>
<tr>
<td>SI-522</td>
<td>Hot leg injection loop 1 check</td>
</tr>
<tr>
<td>SI-523</td>
<td>Hot leg injection loop 1 check</td>
</tr>
<tr>
<td>SI-532</td>
<td>Hot leg injection loop 2 check</td>
</tr>
<tr>
<td>SI-533</td>
<td>Hot leg injection loop 2 check</td>
</tr>
<tr>
<td>SI-540</td>
<td>SI pump #4 discharge check</td>
</tr>
<tr>
<td>SI-541</td>
<td>SI pump #2 discharge check</td>
</tr>
<tr>
<td>SI-542</td>
<td>SI pump #3 discharge check</td>
</tr>
<tr>
<td>SI-543</td>
<td>SI pump #1 discharge check</td>
</tr>
<tr>
<td>SI-618</td>
<td>SI line 4 leakage return</td>
</tr>
<tr>
<td>SI-628</td>
<td>SI line 2 leakage return</td>
</tr>
<tr>
<td>SI-638</td>
<td>SI line 3 leakage return</td>
</tr>
<tr>
<td>SI-648</td>
<td>SI line 1 leakage return</td>
</tr>
<tr>
<td>SI-651</td>
<td>SC pump #1 suction</td>
</tr>
<tr>
<td>SI-652</td>
<td>SC pump #2 suction</td>
</tr>
<tr>
<td>SI-653</td>
<td>SC pump #1 suction</td>
</tr>
<tr>
<td>SI-654</td>
<td>SC pump #2 suction</td>
</tr>
</tbody>
</table>
### Table 3.9-15 (1 of 2)

**Design and Arrangement Comparison of Reactor Internals**

<table>
<thead>
<tr>
<th></th>
<th>Palo Verde</th>
<th>APR1400</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Core Support Barrel</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A.</td>
<td>Right circular cylinder, three sections, supported by heavy external flange top end, heavy internal flange bottom end.</td>
<td>Right circular cylinder, three sections, supported by heavy external flange top end, heavy internal flange bottom end.</td>
</tr>
<tr>
<td>B.</td>
<td>Two outlet nozzles through barrel.</td>
<td>Two outlet nozzles through barrel.</td>
</tr>
<tr>
<td>C.</td>
<td>Top flange seats on reactor vessel ledge, and has four alignment keys attached.</td>
<td>Top flange seats on reactor vessel ledge, and has four alignment keys attached.</td>
</tr>
<tr>
<td>D.</td>
<td>Bottom flange supports lower support structure, fuel, and core shroud.</td>
<td>Bottom flange supports lower support structure, fuel, and core shroud.</td>
</tr>
<tr>
<td>E.</td>
<td>Top flange supports upper guide structure assembly.</td>
<td>Top flange supports upper guide structure assembly.</td>
</tr>
<tr>
<td>F.</td>
<td>Six amplitude-limiting devices (snubbers) attached to lower barrel.</td>
<td>Six amplitude-limiting devices (snubbers) attached to lower barrel.</td>
</tr>
<tr>
<td><strong>Lower Support Structure</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A.</td>
<td>Made up of interlocked grid beams with surrounding short cylinder and perforated bottom plates attached to the bottoms of the beams.</td>
<td>Made up of interlocked grid beams with surrounding short cylinder and perforated bottom plates attached to the bottom of the beams.</td>
</tr>
<tr>
<td>B.</td>
<td>Fuel support pins are attached to the top end of the grid beams.</td>
<td>Fuel support pins are attached to the top end of the grid beams.</td>
</tr>
<tr>
<td>C.</td>
<td>The core shroud assembly is attached to the top of the LSS cylinder.</td>
<td>The core shroud assembly is attached to the top of the LSS cylinder.</td>
</tr>
<tr>
<td>D.</td>
<td>The LSS cylinder rests on and is attached to the CSB bottom (internal) flange.</td>
<td>The LSS cylinder rests on and is attached to the CSB bottom (internal) flange.</td>
</tr>
</tbody>
</table>
### Upper Guide Structure

<table>
<thead>
<tr>
<th>Palo Verde</th>
<th>APR1400</th>
</tr>
</thead>
<tbody>
<tr>
<td>A. Right circular cylinder supported by a heavy external flange type end, and heavy plate attached to bottom end.</td>
<td>A. Right circular cylinder supported by a heavy external flange top end, and heavy plate attached to bottom end.</td>
</tr>
<tr>
<td>B. Heavy plate in A is perforated with flow holes and guide tubes.</td>
<td>B. Heavy plate in A is perforated with flow holes and guide tubes.</td>
</tr>
<tr>
<td>C. A second heavy perforated plate supports the bottom ends of the guide tubes, is also perforated.</td>
<td>C. A second heavy perforated plate supports the bottom ends of the guide tubes, is also perforated.</td>
</tr>
<tr>
<td>D. This second heavy plate in C engages with guide lugs on the core shroud.</td>
<td>D. This second heavy plate in C engages with guide lugs on the core shroud.</td>
</tr>
<tr>
<td>E. The guide tubes are welded to the two heavy plates above.</td>
<td>E. The guide tubes are welded to the two heavy plates above.</td>
</tr>
</tbody>
</table>

### CEA Shroud Assembly / Inner Barrel Assembly

<table>
<thead>
<tr>
<th>Palo Verde</th>
<th>APR1400</th>
</tr>
</thead>
<tbody>
<tr>
<td>A. A series of large-diameter tubes is connected by full-length webs</td>
<td>A. A series of large-diameter tubes is connected by full-length webs.</td>
</tr>
<tr>
<td>B. All welded construction.</td>
<td>B. All welded construction.</td>
</tr>
<tr>
<td>C. N/A</td>
<td>C. The shroud tube and web assembly is connected to an external cylinder.</td>
</tr>
<tr>
<td>D. The tube and web assembly is supported by the UGS support plate via tie-rods. Incorporates four interlocking snubbers to the upper UGS.</td>
<td>D. The tube, web, and cylinder assembly is supported by the UGS upper flange.</td>
</tr>
<tr>
<td>E. Material – austenitic stainless steel</td>
<td>E. Material – austenitic stainless steel</td>
</tr>
</tbody>
</table>
### Nominal Dimensional Comparison of Reactor Internals

<table>
<thead>
<tr>
<th>Component</th>
<th>Palo Verde</th>
<th>APR1400</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Core Support Barrel</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Length, mm (in)</td>
<td>9,734.6 (383-1/4)</td>
<td>9,715.5 (382-1/2)</td>
</tr>
<tr>
<td>Diameter (ID), mm (in)</td>
<td>3,987.8 (157)</td>
<td>3,987.8 (157)</td>
</tr>
<tr>
<td>Thickness upper, mm (in)</td>
<td>76.2 (3)</td>
<td>76.2 (3)</td>
</tr>
<tr>
<td>Thickness middle, mm (in)</td>
<td>66.7 (2-5/8)</td>
<td>66.7 (2-5/8)</td>
</tr>
<tr>
<td>Thickness lower, mm (in)</td>
<td>76.2 (3)</td>
<td>76.2 (3)</td>
</tr>
<tr>
<td>Outlet nozzles (qty)</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>Outlet nozzle diameter (ID), mm (in)</td>
<td>1,184.3 (46-5/8)</td>
<td>1,184.3 (46-5/8)</td>
</tr>
<tr>
<td>Snubbers (qty)</td>
<td>6</td>
<td>6</td>
</tr>
<tr>
<td><strong>Lower Support Structure</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cylinder height, mm (in)</td>
<td>412.8 (16-1/4)</td>
<td>412.8 (16-1/4)</td>
</tr>
<tr>
<td>Cylinder diameter (OD), mm (in)</td>
<td>3,970.3 (156-5/16)</td>
<td>3,970.3 (156-5/16)</td>
</tr>
<tr>
<td>Main beams (qty)</td>
<td>16</td>
<td>16</td>
</tr>
<tr>
<td>Beam thickness, mm (in)</td>
<td>44.5 (1-3/4)</td>
<td>44.5 (1-3/4)</td>
</tr>
<tr>
<td>Beam height, mm (in)</td>
<td>669.9 (26-3/8)</td>
<td>669.9 (26-3/8)</td>
</tr>
<tr>
<td><strong>UGS Support Barrel Assembly</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Length, mm (in)</td>
<td>4,924.4 (193-7/8)</td>
<td>4,924.4 (193-7/8)</td>
</tr>
<tr>
<td>Diameter flange (OD), mm (in)</td>
<td>4,559.3 (179-1/2)</td>
<td>4,559.3 (179-1/2)</td>
</tr>
<tr>
<td>Diameter barrel (OD), mm (in)</td>
<td>3,962.4 (156)</td>
<td>3,962.4 (156)</td>
</tr>
<tr>
<td>Barrel thickness, mm (in)</td>
<td>76.2 (3)</td>
<td>76.2 (3)</td>
</tr>
<tr>
<td>CEA guide tubes (qty)</td>
<td>804</td>
<td>820</td>
</tr>
<tr>
<td>Fuel alignment plate diameter, mm (in)</td>
<td>3,962.4 (156)</td>
<td>3,962.4 (156)</td>
</tr>
<tr>
<td>Plate thickness, mm (in)</td>
<td>114.3 (4-1/2)</td>
<td>139.7 (5-1/2)</td>
</tr>
</tbody>
</table>
### Table 3.9-17

Comparison of Operating Condition for Reactor Internals

<table>
<thead>
<tr>
<th>Operating Condition</th>
<th>Palo Verde</th>
<th>APR1400</th>
</tr>
</thead>
<tbody>
<tr>
<td>RV design pressure, kg/cm²A (psia)</td>
<td>175.8 (2,500)</td>
<td>175.8 (2,500)</td>
</tr>
<tr>
<td>RV normal operating pressure, kg/cm²A (psia)</td>
<td>158.2 (2,250)</td>
<td>158.2 (2,250)</td>
</tr>
<tr>
<td>Normal operating coolant inlet temperature, °C (°F)</td>
<td>296.1 (565)</td>
<td>290.6 (555)</td>
</tr>
<tr>
<td>Normal operating coolant outlet temperature, °C (°F)</td>
<td>327.3 (621.2)</td>
<td>323.9 (615)</td>
</tr>
<tr>
<td>Design temperature, °C (°F)</td>
<td>343.3 (650)</td>
<td>343.3 (650)</td>
</tr>
<tr>
<td>Normal operating coolant flow rate, L/min (gpm)</td>
<td>1,687,000 (445,600)</td>
<td>1,689,000 (446,300)</td>
</tr>
<tr>
<td>Mechanical design coolant flow rate (with core), L/min (gpm)</td>
<td>1,965,000 (519,124)</td>
<td>1,943,000 (513,200)</td>
</tr>
<tr>
<td>Mechanical design coolant flow rate (without core), L/min (gpm)</td>
<td>2,066,000 (545,860)</td>
<td>2,112,000 (557,900)</td>
</tr>
<tr>
<td>Reactor coolant pump frequency (Hz)</td>
<td>20</td>
<td>20</td>
</tr>
<tr>
<td>Reactor coolant pump blade passing frequency (Hz)</td>
<td>120</td>
<td>120</td>
</tr>
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</table>
## APR1400 DCD TIER 2

Table 3.9-18 (1 of 2)

List of Active Pumps

<table>
<thead>
<tr>
<th>Pump</th>
<th>ASME Class</th>
<th>System (Subsection)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety injection pump 1</td>
<td>2</td>
<td>6.3.2</td>
</tr>
<tr>
<td>Safety injection pump 2</td>
<td>2</td>
<td>6.3.2</td>
</tr>
<tr>
<td>Safety injection pump 3</td>
<td>2</td>
<td>6.3.2</td>
</tr>
<tr>
<td>Safety injection pump 4</td>
<td>2</td>
<td>6.3.2</td>
</tr>
<tr>
<td>Shutdown cooling pump 1</td>
<td>2</td>
<td>6.3.2</td>
</tr>
<tr>
<td>Shutdown cooling pump 2</td>
<td>2</td>
<td>6.3.2</td>
</tr>
<tr>
<td>SFP cooling pump 1</td>
<td>3</td>
<td>9.1.3</td>
</tr>
<tr>
<td>SFP cooling pump 2</td>
<td>3</td>
<td>9.1.3</td>
</tr>
<tr>
<td>CCW pump 1A</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>CCW pump 1B</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>CCW pump 2A</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>CCW pump 2B</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>CCW makeup pump 3A</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>CCW makeup pump 3B</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>ESW pump 1A</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>ESW pump 1B</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>ESW pump 2A</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>ESW pump 2B</td>
<td>3</td>
<td>9.2.2</td>
</tr>
<tr>
<td>ECW pump PP01A</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>ECW pump PP02A</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>ECW makeup pump PP03A</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>ECW pump PP01B</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>ECW pump PP02B</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>ECW makeup pump PP03B</td>
<td>3</td>
<td>9.2.7</td>
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### Table 3.9-18 (2 of 2)

<table>
<thead>
<tr>
<th>Pump</th>
<th>ASME Class</th>
<th>System (subsection)</th>
</tr>
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<tbody>
<tr>
<td>Essential chilled water pump PP01A</td>
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<td>9.2.7</td>
</tr>
<tr>
<td>Essential chilled water pump PP02A</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>Essential chilled water makeup pump PP03A</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>Essential chilled water pump PP01B</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>Essential chilled water pump PP02B</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>Essential chilled water makeup pump PP03B</td>
<td>3</td>
<td>9.2.7</td>
</tr>
<tr>
<td>DG 1 fuel oil transfer pump 01A/02A</td>
<td>3</td>
<td>9.5.4</td>
</tr>
<tr>
<td>DG 1 fuel oil transfer pump 01B/02B</td>
<td>3</td>
<td>9.5.4</td>
</tr>
<tr>
<td>MD AFW pump PP02A</td>
<td>3</td>
<td>10.4.9</td>
</tr>
<tr>
<td>TD AFW pump PP01A</td>
<td>3</td>
<td>10.4.9</td>
</tr>
<tr>
<td>MD AFW pump PP02B</td>
<td>3</td>
<td>10.4.9</td>
</tr>
<tr>
<td>TD AFW pump PP01B</td>
<td>3</td>
<td>10.4.9</td>
</tr>
</tbody>
</table>
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Figure 3.9-2  Summary of Analytical Methodology
Figure 3.9-3  ASHSD Finite Element Model of Core Support Barrel
Figure 3.9-4  Finite Element Model of Inner Barrel Assembly
Figure 3.9-5  Finite Element Model of the CEA Guide Tubes
Figure 3.9-6  Finite Element Model of Lower Support Structure / In-Core Instrumentation Nozzle Assembly
RSPT Assembly
Upper Shroud
Coil Stack Assembly
Upper Lift Coil
Upper Latch Coil
Upper Latch
Upper Pressure Housing Assembly
Omega Seal
Motor Housing Assembly
Motor Assembly
Lower Lift Coil
Lower Latch Coil
Lower Latch
Omega Seal
Reacto Vessel Closure Head Nozzle
Housing Nut
Vent Stem
Magnet Assembly
Upper Pressure Housing Assembly
Omega Seal
Motor Assembly
Pressure Boundary

Figure 3.9-7  Control Element Drive Mechanism
Figure 3.9-8 Reactor Internals Arrangement
Figure 3.9-9  Core Support Barrel Assembly
Figure 3.9-10  Reactor Vessel / Core Support Barrel Snubber Assembly
Figure 3.9-11  Lower Support Structure / ICI Nozzle Assembly
Figure 3.9-12 Core Shroud
Figure 3.9-13  Upper Guide Structure Assembly
Figure 3.9-14  In-Core Instrumentation Support System
Figure 3.9-15  Horizontal Seismic and Pipe Break Analysis Model of Reactor Internals
Figure 3.9-16 Core Seismic Analysis Model – One Row of 17 Fuel Assemblies
Figure 3.9-17  Vertical Seismic and Pipe Break Analysis Model of Reactor Internals
APPENDIX 3.9A

SUPPLEMENTAL INFORMATION ON CRITERIA
OF THE APR1400 DISTRIBUTION SYSTEMS
# APPENDIX 3.9A – SUPPLEMENTAL INFORMATION ON CRITERIA OF THE
# APR1400 DISTRIBUTION SYSTEMS

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<th>TITLE</th>
<th>PAGE</th>
</tr>
</thead>
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<td>HVAC Ductwork and Supports</td>
<td>3.9A-1</td>
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<td>General</td>
<td>3.9A-1</td>
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<tr>
<td>3.9A.1.2</td>
<td>Design Considerations</td>
<td>3.9A-1</td>
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<td>3.9A.1.2.1</td>
<td>Internal Pressure Load ($P_0$, $P_A$)</td>
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<td>3.9A.1.2.2</td>
<td>Dead Load (D)</td>
<td>3.9A-1</td>
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<td>3.9A.1.2.3</td>
<td>Thermal Load ($T_0$, $T_A$)</td>
<td>3.9A-1</td>
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<td>3.9A.1.2.4</td>
<td>Seismic Load (SSE)</td>
<td>3.9A-2</td>
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<td>3.9A.1.2.5</td>
<td>Live Load (L)</td>
<td>3.9A-2</td>
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<td>3.9A.1.2.6</td>
<td>External Pressure Differential (EPD)</td>
<td>3.9A-2</td>
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<td>3.9A.1.2.7</td>
<td>Wind Load (W)</td>
<td>3.9A-2</td>
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<td>3.9A.1.3</td>
<td>Load Combinations</td>
<td>3.9A-2</td>
</tr>
<tr>
<td>3.9A.1.4</td>
<td>Analysis and Acceptance Criteria</td>
<td>3.9A-3</td>
</tr>
<tr>
<td>3.9A.1.4.1</td>
<td>General</td>
<td>3.9A-3</td>
</tr>
<tr>
<td>3.9A.1.4.2</td>
<td>Damping Values</td>
<td>3.9A-3</td>
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<td>3.9A.1.4.3</td>
<td>Seismic Analysis</td>
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</tr>
<tr>
<td>3.9A.1.5</td>
<td>Allowable Stress Criteria</td>
<td>3.9A-4</td>
</tr>
<tr>
<td>3.9A.1.5.1</td>
<td>Basic Allowable Stress</td>
<td>3.9A-4</td>
</tr>
<tr>
<td>3.9A.1.6</td>
<td>Allowable Deflection Criteria</td>
<td>3.9A-5</td>
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<td>3.9A.1.7</td>
<td>Welding and Weld Acceptance Criteria</td>
<td>3.9A-5</td>
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<td>3.9A-5</td>
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<td>3.9A-5</td>
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<td>3.9A.2.2</td>
<td>Design Considerations</td>
<td>3.9A-5</td>
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<tr>
<td>3.9A.2.2.1</td>
<td>Dead Load (D)</td>
<td>3.9A-5</td>
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<tr>
<td>3.9A.2.2.2</td>
<td>Live Load (L)</td>
<td>3.9A-6</td>
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</tbody>
</table>
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APPENDIX 3.9A – SUPPLEMENTAL INFORMATION ON CRITERIA OF THE APR1400 DISTRIBUTION SYSTEMS

3.9A.1 HVAC Ductwork and Supports

3.9A.1.1 General

Heating, ventilation, and air conditioning (HVAC) ductwork is designed and supported to withstand the loading combinations presented in this section, as applicable. The design and analysis guidelines herein apply to seismic Category I and II HVAC ductwork and supports. Seismic Category II HVAC ductwork and supports, as defined in Subsection 3.2.1, are analyzed to provide reasonable assurance that their failure would not adversely impact safety-related equipment or components. The SSCs design procedure for differential settlement and relative displacement is described in Subsection 3.8.5.8.

3.9A.1.2 Design Considerations

3.9A.1.2.1 Internal Pressure Load ($P_o, P_A$)

Internal pressure loads ($P_o, P_A$) do not affect the design of HVAC duct supports but should be considered in the design of ductwork. Sheet Metal and Air Conditioning Contractors’ National Association (SMACNA) guidelines (References 1 and 2) are used in determining duct thickness, stiffener, and companion angle requirements.

3.9A.1.2.2 Dead Load (D)

Dead load (D) includes the weight of the ductwork itself, in-line components (e.g., dampers, humidifiers, in-duct electric heaters), externally mounted components, and insulation. Self-weight of structural members and dead load of ductwork are considered in the design of HVAC duct supports. An additional 23 kg (50 lb) concentrated load is considered in the design of HVAC duct supports for attachments such as conduits and lighting fixtures.

3.9A.1.2.3 Thermal Load ($T_o, T_A$)

Stresses resulting from the movement of supports or expansion of ductwork under temperature changes are avoided by using expansion joints in the system design. For ducts with gasket companion angles, thermal loads are negligible.
3.9A.1.2.4  **Seismic Load (SSE)**

Seismic loads on HVAC ductwork and support systems are considered using the methods described in Subsection 3.9A.1.4.3. Stresses are determined from the seismic excitation in each of the three orthogonal directions by a square root of the sum of squares (SRSS) method.

3.9A.1.2.5  **Live Load (L)**

HVAC duct supports are designed to withstand the expected live load. The live load considered is a construction/maintenance man-load of 114 kg (250 lb). Live load is not considered in the design of HVAC ductwork.

3.9A.1.2.6  **External Pressure Differential (EPD)**

HVAC ductwork and supports are designed to withstand dynamic external pressure differential (EPD) loads resulting from postulated pipe breaks. This condition is normally precluded by routing ductwork away from the affected area.

3.9A.1.2.7  **Wind Load (W)**

Exposed safety-related HVAC ductwork and supports are designed to withstand forces generated by wind.

3.9A.1.3  **Load Combinations**

The load combinations to be considered for the design of HVAC ductwork and support are as follows:

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Load Combination</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (Normal)</td>
<td>$D + L + P_o + T_o$</td>
</tr>
<tr>
<td>B (Severe)</td>
<td>$D + W + P_o + T_o$</td>
</tr>
<tr>
<td>C (Extreme)</td>
<td>$D + SSE + P_A + T_A$</td>
</tr>
<tr>
<td>D (Abnormal)</td>
<td>$D + SSE + EPD + P_A + T_A$</td>
</tr>
</tbody>
</table>

Where:

- $D$ = dead load
- $L$ = live load
- $W$ = wind load
3.9A.1.4 Analysis and Acceptance Criteria

3.9A.1.4.1 General

Structural integrity of HVAC ductwork and supports is demonstrated to provide reasonable assurance that the ductwork and support functions are not impaired. Ductwork stresses are maintained within the allowable limits. Ductwork deflection is limited to the maximum deflection criteria. HVAC duct support stresses are also maintained within the allowable limits.

3.9A.1.4.2 Damping Values

The following damping values are used in the design of HVAC ductwork and supports:

<table>
<thead>
<tr>
<th>Structures</th>
<th>Damping Values</th>
</tr>
</thead>
<tbody>
<tr>
<td>HVAC ductwork</td>
<td>7%</td>
</tr>
<tr>
<td>HVAC duct supports</td>
<td>4%</td>
</tr>
</tbody>
</table>

3.9A.1.4.3 Seismic Analysis

3.9A.1.4.3.1 Equivalent Static Analysis Method

The equivalent static analysis method is used for HVAC ductwork and supports. The system response is assumed to be the peak of the required response spectra. This response is then multiplied by a static coefficient of 1.5. The seismic load in the design of HVAC ductwork and supports is obtained by multiplying the peak acceleration by a static coefficient of 1.5 and the participating mass.
3.9A.1.4.3.2 Dynamic Analysis Method

If a specific dynamic analysis is required, HVAC ductwork and supports are modeled to accurately represent the mass distribution and stiffness characteristics. The response spectrum modal analysis or time history analysis can be applicable.

3.9A.1.5 Allowable Stress Criteria

All HVAC ductwork and supports safely sustain stresses induced by the various loading conditions. The stress values are provided as a basis for evaluating the required structural integrity of the HVAC ductwork and supports.

For HVAC ductwork, the allowable stresses for the various operating conditions are as follows:

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Allowable Stress</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (Normal)</td>
<td>Basic Allowable Stress</td>
</tr>
<tr>
<td>B (Severe)</td>
<td>Basic Allowable Stress</td>
</tr>
<tr>
<td>C (Extreme)</td>
<td>$1.6 \times$ Basic Allowable Stress</td>
</tr>
<tr>
<td>D (Abnormal)</td>
<td>$1.7 \times$ Basic Allowable Stress</td>
</tr>
</tbody>
</table>

For stiffeners, companion angles, and HVAC duct supports, the allowable stresses for the various operating conditions are as follows:

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Allowable Stress</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (Normal)</td>
<td>Basic Allowable Stress</td>
</tr>
<tr>
<td>B (Severe)</td>
<td>Basic Allowable Stress</td>
</tr>
<tr>
<td>C (Extreme)</td>
<td>$1.6 \times$ Basic Allowable Stress</td>
</tr>
</tbody>
</table>

Connections are designed so that stress levels in welds and bolts do not exceed the basic allowable stresses under any of the operating conditions.

3.9A.1.5.1 Basic Allowable Stress

3.9A.1.5.1.1 HVAC Ductwork

Basic allowable stresses for HVAC ductwork are per the American Iron and Steel Institute (AISI) “Cold-Formed Steel Design Manual” (Reference 3). The basic allowable stress for the stiffener, companion angle for HVAC ductwork is determined based on ANSI/AISC

3.9A.1.5.1.2 HVAC Duct Supports

Basic allowable stresses for structural steel, welds, and bolts are per ANSI/AISC N690 (Reference 4).

3.9A.1.6 Allowable Deflection Criteria

Maximum deflection is evaluated to provide reasonable assurance that the duct function is not impaired and the required clearances are maintained.

3.9A.1.7 Welding and Weld Acceptance Criteria

Welding activities for HVAC ductwork are accomplished in accordance with American Welding Society (AWS) D1.3 (Reference 5). For HVAC duct supports, welding activities are fulfilled in accordance with AWS D1.1 (Reference 6). The visual acceptance criteria of welding are defined in standard NCIG-01 (Reference 7).

3.9A.2 Cable Tray/Conduit Supports

3.9A.2.1 General

Cable tray and conduit are designed and supported to withstand the loading combinations presented in this section, as applicable. The design and analysis guidelines herein apply to seismic Category I and II cable tray/conduit supports as defined in Subsection 3.2.1. The SSCs design procedure for differential settlement and relative displacement is described in Subsection 3.8.5.8.

3.9A.2.2 Design Considerations

3.9A.2.2.1 Dead Load (D)

Dead loads (D) include the weight of the cable tray or conduit, fittings, covers, and any other dead loads applied to the system. An additional 23 kg (50 lb) concentrated load is considered in the design of the cable tray supports for attachments such as conduits and lighting fixtures.
3.9A.2.2.2 **Live Load (L)**

Cable tray supports are designed to withstand expected live load. The live load considered is a construction/maintenance man load of 90 kg (200 lb). Live load is not considered in the design of conduit supports. However, where live loads such as wind and snow are applicable, they are considered in the design.

3.9A.2.2.3 **Seismic Load (SSE)**

Cable tray/conduit supports are designed to withstand seismic load. Stresses are determined for the seismic excitation in each of the three orthogonal directions by the SRSS method. Seismic load is determined using the equivalent static analysis method or the dynamic analysis method.

3.9A.2.2.3.1 **Equivalent Static Analysis Method**

The equivalent static analysis method is used for cable tray/conduit supports. The system response is assumed to be the peak of the required response spectra. This response is then multiplied by a static coefficient of 1.5. The seismic load in the design of the cable tray/conduit supports is obtained by multiplying the peak acceleration by a static coefficient of 1.5 and the participating mass.

3.9A.2.2.3.2 **Dynamic Analysis Method**

If specific dynamic analysis is required, the cable tray/conduit supports are modeled to accurately represent their mass distribution and stiffness characteristics. The response spectrum modal analysis or time-history analysis can be used.

3.9A.2.3 **Load Combinations and Allowable Stress Criteria**

The loading combinations considered for the design of cable tray/conduit supports are as follows:

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Load Combination</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (Normal)</td>
<td>D + L</td>
</tr>
<tr>
<td>C (Extreme)</td>
<td>D + SSE</td>
</tr>
</tbody>
</table>

Where:

\[ D = \text{dead load} \]
L = live load (only for cable tray supports)
SSE = safe shutdown earthquake

For cable tray/conduit supports, the allowable stresses for the various operating conditions are as follows:

<table>
<thead>
<tr>
<th>Service Level</th>
<th>Allowable Stress</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (Normal)</td>
<td>Basic Allowable Stress</td>
</tr>
<tr>
<td>C (Extreme)</td>
<td>1.6 × Basic Allowable Stress</td>
</tr>
</tbody>
</table>

3.9A.2.3.1 Basic Allowable Stress

Basic allowable stresses for the structural steel, welds, and bolts of cable tray/conduit supports are determined based on ANSI/AISC N690 (Reference 4).

3.9A.2.4 Damping Value

The following damping values are used in the design of the cable tray/conduit supports:

<table>
<thead>
<tr>
<th>Structures</th>
<th>Damping Values</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cable tray support</td>
<td>7%</td>
</tr>
<tr>
<td>Conduit support</td>
<td>5%</td>
</tr>
</tbody>
</table>

3.9A.2.5 Welding and Weld Acceptance Criteria

Welding activities for cable tray/conduit supports are accomplished in accordance with AWS D1.1 (Reference 6). The visual acceptance criteria of welding are defined in standard NCIG-01 (Reference 7).

3.9A.3 References


APPENDIX 3.9B

REACTOR COOLANT SYSTEM ANALYSIS
# APPENDIX 3.9B – REACTOR COOLANT SYSTEM ANALYSIS

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<td>Reactor Coolant System Structural Analysis Model</td>
<td>3.9B-9</td>
</tr>
<tr>
<td>Figure 3.9B-2</td>
<td>Pressurizer Structural Analysis Model</td>
<td>3.9B-10</td>
</tr>
</tbody>
</table>
3.9B.1 Introduction

This appendix describes the methods that are used to analyze the reactor coolant system (RCS).

The RCS has two loops that are connected to the reactor vessel (RV). Each loop consists of one steam generator (SG), two reactor coolant pumps (RCPs), one hot leg pipe connecting RV and SG, and two cold leg pipes connecting SG, RCP, and RV. One pressurizer (PZR) is connected to one of the hot leg pipes. All components are located inside the containment building. The arrangement of the RCS is shown in Figures 5.1.3-1 and 5.1.3-2.

The RCS structural analyses for safe shutdown earthquake (SSE) and in-containment refueling water storage tank (IRWST) discharge events are performed using the coupled model of the RCS, PZR, and containment building. The structural analyses for normal operating conditions and branch line pipe breaks (BLPBs) are performed by using separate RCS and the PZR models with the building stiffnesses at the support interfaces.

Dynamic analyses of the RCS and PZR under SSE, BLPB, and IRWST discharge conditions are performed by using time-history analysis methods. A time-history analysis adopts direct integration, mode superposition, and complex frequency response method.

The results of the static and dynamic analyses are used for the design and analysis of the RCS components and their substructures. The major components of the RCS are designed in accordance with ASME Section III (Reference 1). Reasonable assurance of the structural integrity is provided by meeting the stress and fatigue limits in ASME Section III (Reference 1).

3.9B.2 Reactor Coolant System Structural Model

This section describes the structural analysis models of the RCS and PZR. The models are created using the finite element analysis code ANSYS, which is described in Subsections 3.9.1.2.1.6 and 3.9.1.2.2.1.
The two-loop RCS structural model, a lumped-mass stick (or beam) model, consists of the representations of the RCS components, RCS supports and reactor coolant loop (RCL) piping. The component models consist of one RV with its internals, including the fuel and the supports; two SGs with their internals and supports; and four RCPs with motors and supports. All branch pipelines are eliminated from the RCS model because the mass and rigidity of the piping do not significantly influence the dynamic behavior of the RCS. This dynamic decoupling is in accordance with the decoupling criteria for the seismic analysis specified in NUREG-0800, SRP 3.7.2 (Reference 2).

The RCS structure is mathematically represented by the elements in the ANSYS library listed below. The model is developed using the nominal dimensions and locations. The RCS and PZR models are illustrated in Figures 3.9B-1 and 3.9B-2, respectively. The spatial locations and orientations are defined by the set of orthogonal axes in which the y axis represents the vertical direction, and the x and z axes are in the horizontal plane.

The ANSYS library elements are as follows:

a. Beam element (3-dimensional)
b. Pipe element (3-dimensional straight pipe and elbow)
c. Spar element (3-dimensional)
d. Stiffness matrix element (3-dimensional)
e. Lumped mass (3-dimensional)

The beam element properties include the cross-sectional area, the moments of inertia about the x, y, and z axes of the element coordinate system, which is dependent on element orientation, and the shear areas. The beam element represents the portions with basic cross-sectional shapes such as a circle or rectangle. Typically, the RCS components and RV vertical support columns are represented by beam elements.

The straight pipe element properties are the inner and outer diameters, and elbow pipe element properties include the inner and outer diameters and the radius of curvature with
options for in-plane and out-of-plane bending flexibility factors. The pipe element represents primarily the RCL piping.

The spar element property is the cross-sectional area. The spar element represents primarily the supports pinned at both ends such as RCP support columns and the SG and RCP snubbers.

The stiffness matrix element represents the portions with complicated features that are not simply represented by beam or pipe elements. The stiffnesses of the RV primary inlet and outlet nozzles, and key attachments to components, are determined from static analyses using detailed finite element models, and the stiffnesses are modeled using the matrix elements.

The mass of each component and its enclosed fluid is distributed where the dynamic characteristics can be represented and where the dynamic characteristic of the components can be adequately transmitted to other components by considering possible dynamic interactions between the components. The lumped masses are distributed on the locations that can maintain the center of gravity and the mass moment of inertia of the components.

The PZR is modeled in the same way the RCS model is generated and with the same element types and mass discretization.

To account for dynamic interactions between the RV and the integrated head assembly (IHA) including the control element driving mechanisms (CEDMs), the IHA reduced model is coupled to the RV model. The IHA reduced model is generated in the same way the RCS model is generated.

3.9B.2.1 RCS Component Supports

a. Reactor supports

The RV is supported by four vertical columns located under the vessel inlet nozzles. A column pad integrally attached to the vessel inlet nozzles is placed in the horizontal direction to allow radial growth of the vessel during thermal expansion and to restrain tangential motions for dynamic load conditions. Four
vertical columns also support the RV in the vertical direction. The supports are designed to accommodate normal operation, seismic, IRWST discharge, and BLPB loads. The column base plate acts as a keyway to restrain the bottom of the RV for dynamic load conditions. Typical RV supports are shown in Figure 3.8-15.

b. Steam generator supports

The SG is supported at the bottom by a sliding base bolted to an integrally attached conical skirt. The sliding base rests on low-friction spherical head bearings, which allow unrestrained thermal expansion of the RCS. Two keyways in the sliding base mate with embedded keys to guide the movement of the SG during expansion and contraction of the RCS and also limit movement of the bottom of the SG during seismic, IRWST discharge, and BLPB events.

A system of keys and snubbers located on the steam drum guides the top of the SG during thermal expansion and contraction of the RCS and provide support during seismic, IRWST discharge, and BLPB events. Typical SG supports are shown in Figure 3.8-16.

c. Reactor coolant pump supports

RCP supports consist of four vertical columns that support the vertical loads of the RCP, two horizontal snubbers, two upper horizontal columns, and two lower horizontal columns. The vertical and horizontal columns provide support for the pumps during normal operation, seismic, IRWST discharge, and BLPB conditions. Typical RCP supports are shown in Figure 3.8-17.

d. Pressurizer supports

The pressurizer is supported by a cylindrical skirt, as shown in Figure 3.8-18. This skirt is welded to the pressurizer and bolted to the support structure. The skirt is designed to withstand deadweight and normal operating loads as well as the loads due to seismic, pressurizer pilot-operated safety relief valve (POSRV) actuation, IRWST discharge, and BLPB events. Four keys welded to the upper
shell of the pressurizer provide restraints for seismic, pressurizer POSRV actuation, IRWST discharge, and BLPB events.

3.9B.3 Static Analysis

To determine the structural behaviors for normal operating conditions, static analyses are performed. The RCS and the PZR is statically analyzed for the deadweight, internal pressure, and thermal expansion.

For the static analyses, special considerations are given to the supports and the branch lines: the snubbers and gapped supports are released to avoid restraining the RCS and the PZR, and the reaction loads determined from the branch line analyses at the nozzles on the components and piping are imposed on the corresponding nozzles to account for the influences of the branch lines.

3.9B.4 Dynamic Analysis

The RCS is designed to withstand the combined effects of normal operating conditions together with the dynamic loads such as earthquakes, postulated pipe breaks, and IRWST discharge events. Dynamic analyses for these three events are performed to determine the dynamic responses of the RCS and the PZR. This section describes the dynamic analyses.

Dynamic analyses are performed using the time-history method of dynamic response analysis.

The damping values used in the analysis of seismic Category I and II structures, systems, and components are selected from Table 3.7-7. The damping values given in Table 3.7-7 include those recommended in NRC RG 1.61, (Reference 3).

The results of dynamic analyses contain the forces and moments, maximum displacements, response spectra, and time histories. The results of the RCS dynamic analyses are used for the design and analysis of RCS components and substructures including connected branch lines. The resultant response spectra are broadened by 15 percent to account for uncertainties in the frequencies of the structural model.
3.9B.4.1 Seismic Analysis

The RCS seismic model is coupled with the finite element model of the containment internal structures, which is incorporated into the model of nuclear island structures, as described in Subsection 3.7.2.3.3. The RCS model consists of the RV, SG, RCP, RCL piping, PZR, and PZR surge line. The RCS and the PZR models are described in Section 3.9B.2 and shown in Figures 3.9B-1 and 3.9B-2, respectively.

As described in Subsection 3.7.2.4, the soil-structure interaction (SSI) analysis of seismic Category I structures is performed using the complex frequency response method. The model of nuclear island structures including the RCS is used in the SSI analysis. The seismic responses of the RCS are determined from the SSI analysis. Earthquake input motion for the SSI analysis in the form of synthetic acceleration time histories is described in Subsection 3.7.1.1.2.

3.9B.4.2 Postulated Pipe Break Analysis

To determine the structural responses to the break effects of the pipelines to which the leak-before-break (LBB) concept is not applied, the structural analyses of the RCS and the PZR are performed for each break. For the analyses, time-dependent break effects are applied to the RCS and the PZR models, which are modified from the model described in Section 3.9B.2 to accommodate the various dynamic effects: the mass is distributed to more nodes, and the gaps in the support systems are modeled. With the geometric nonlinearities of the gaps, the analyses are performed using the nonlinear time-history analysis method. The integration time step for the analyses is short enough to be able to consider the instantaneous dynamic effects of pipe breaks. Break effects of a specific break are applied to the structural analyses on a case-by-case basis.

Break effects are as follows:

a. Jet impingement and thrust

The determination of pipe thrust and jet impingement loads for postulated pipe breaks is described in Subsection 3.6.2.3.2.1.
b. Subcompartment pressurization

The differential pressurization across the component described in Subsection 6.2.1.2 is considered to be external forces on the components for the structural analyses of the RCS and PZR. External forces resulting from the differential pressurization across the component in a compartment are determined by multiplying the differential pressures in nodalized spaces by the pressurized areas of the components and a factor of 1.4, as described in Subsection 6.2.1.2.3.

c. Blowdown loads

Blowdown loads for postulated pipe breaks are described in Subsection 3.9.2.5.2.

d. Nozzle loads

Nozzle loads imposed by the dynamic motions of the pipe in an intermediate break and a nozzle break when multiple nozzles are connected to the same pipeline are considered for the structural analyses of the RCS and PZR. The loads are determined from the analyses of the piping systems.

e. Blast wave loads

Blast wave loads for postulated pipe breaks are described in Subsection 3.6.2.4.4.

3.9B.4.3 In-Containment Refueling Water Storage Tank Discharge Analysis

The hydrodynamic loads on the IRWST described in Subsection 6.8.4.3 are taken into consideration for the RCS and PZR structural analyses.

The IRWST discharge analysis model is a coupled model of the RCS and the containment internal structures. Time-history analyses are performed to determine structural responses to the hydrodynamic loads on the IRWST.
3.9B.5 References


Figure 3.9B-1 Reactor Coolant System Structural Analysis Model
Figure 3.9B-2 Pressurizer Structural Analysis Model

<table>
<thead>
<tr>
<th>MASS POINT NUMBER</th>
<th>DEGREE OF FREEDOM</th>
</tr>
</thead>
<tbody>
<tr>
<td>6110, 6130</td>
<td>X, Z</td>
</tr>
<tr>
<td>6120</td>
<td>X, Y, Z</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SUPPORT POINT NUMBER</th>
<th>RESTRAINT DIRECTION</th>
</tr>
</thead>
<tbody>
<tr>
<td>7200, 7400</td>
<td>Z</td>
</tr>
<tr>
<td>7300, 7500</td>
<td>X</td>
</tr>
<tr>
<td>6000</td>
<td>FIXED</td>
</tr>
</tbody>
</table>
3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

This section describes the acceptance criteria, code and standards, procedures, and methods applied to the seismic and dynamic qualification of mechanical and electrical equipment including instrumentation to provide reasonable assurance that they will withstand the effects of postulated events and accidents and still be capable of performing their safety-related functions under the full range of normal, transient, seismic, and accident loadings.

Safety-related equipment is the equipment necessary to provide reasonable assurance of the following:

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe shutdown condition

c. The capability to prevent or mitigate the consequences of accidents that would result in potential offsite exposure in excess of the limits stated in 10 CFR Part 100

Seismic Category I equipment includes safety-related equipment and non-safety-related equipment. The seismic Category I equipment covered by this qualification section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment reactor heat removal; equipment essential to preventing significant release of radioactive material to the environment; and instrumentation needed to assess plant and environs conditions during and after an accident as described in NRC RG 1.97 (Reference 1).

The function of this equipment may include:

a. Equipment that performs the above functions automatically

b. Equipment that operators use to perform the above functions manually

c. Equipment for which failure can prevent satisfactory accomplishment of one or more of the above safety functions
This includes equipment in the reactor protection system (RPS), the engineered safety features (ESF) Class 1E equipment, the emergency power system, and all auxiliary safety-related systems and supports.

Examples of mechanical equipment are valves, pumps, fans, and heat exchangers. Examples of electrical and I&C equipment are motor control centers, load centers, battery racks, and indicators. This equipment is divided into two categories:

a. **Active Equipment** – The equipment that must remain functional both during and after all postulated dynamic event such as fans, pumps, valves, motors, switches, relays, and transmitters.

b. **Passive Equipment** – The equipment whose safety-related function does not involve operability but that does require assurance of its structural and pressure integrity both during and after postulated dynamic events. Examples of some common items of equipment classified as passive include tanks, heat exchangers, and filter cabinets.

Seismic Category I SSCs are identified in Table 3.2-1. Safety-related mechanical and electrical equipment including instrumentation is designed to meet seismic Category I requirements to provide reasonable assurance of the ability to initiate required protective actions and to supply power to components required to mitigate the consequences of events that require safety system operation during and after a safe shutdown earthquake (SSE).

Some non-safety-related equipment are identified to meet seismic Category I requirements in Table 3.2-1. These equipment shall be capable of providing their intended function and maintaining structural integrity in accordance with specific design requirement. Fire protection system equipment can be in such category.

Mechanical and electrical equipment including instrumentation designated as seismic Category II is shown to maintain its structural integrity and not adversely impact safety-related equipment during an SSE and during all static and dynamic loads from normal, transient, and accident conditions.

### 3.10.1 Seismic Qualification Criteria

The seismic and dynamic qualification of mechanical and electrical equipment demonstrates the safety system equipment's ability to perform its required function during
and/or after the time it is subjected to the forces resulting from SSE and other related
dynamic loads at the location of the equipment. For the purpose of equipment seismic
qualification, the equipment is qualified with five one-half SSEs followed by one full SSE.

The SSE term is applicable to the site-independent earthquake or the site-specific
earthquake. The expression “safe shutdown earthquake” used for seismic qualification of
SSCs in this section refers to equipment qualified for the site-specific design. The seismic
Category I SSCs are designed for the SSE. Since the operating basis earthquake (OBE) is
defined as one-third the SSE, an explicit analysis or design of the seismic Category I SSCs
based on OBE is not required in accordance with Appendix S of 10 CFR Part 50 (Reference
2) as defined in Subsection 3.7.1.

When the OBE is defined as less than or equal to 1/3 SSE, explicit design or analysis is not
required for the OBE. The COL applicant is to provide documentation that the designs of
seismic Category I SSCs are analyzed for OBE, if OBE is higher than 1/3 SSE (COL
3.10(1)).

3.10.1.1 Qualification Codes and Standards

The seismic and dynamic qualification program provides reasonable assurance that
equipment classified as seismic Category I meets functional performance requirements, as
defined in design specification order of the equipment, during and after dynamic loadings
due to normal operating, transient, seismic, and accident conditions.

The employed methods for the dynamic qualification described in Subsection 3.10.2 are in
accordance with the requirements of GDC 1, GDC 2, GDC 4, GDC 14, GDC 30, and 10
CFR Part 50, Appendix S and Appendix B. The seismic and dynamic testing portion of
the qualification program is performed in a sequence consistent with the requirements of
Section 6 of IEEE Std. 323 (Reference 3).

The recommended guidance and requirements in NRC RG 1.100 (Reference 4) and IEEE
Std. 344 (Reference 5) are used for the development and implementation of methods and
procedures for seismic qualification of mechanical and electrical equipment.

The methods and procedures for seismic qualification of mechanical and electrical
equipment are in accordance with the recommended guidance and requirements in NRC
RG 1.100 and IEEE Std. 344. The seismic or dynamic qualification based on test,
analysis, or a combination of test and analysis are performed except an experience-based
qualification. An experience-based qualification is not used for any equipment until it is endorsed by NRC RG 1.100.

The seismic Category I mechanical equipment is designed to provide reasonable assurance of structural integrity of pressure boundary components for the intended service load conditions identified in the equipment’s design specification, in accordance with the requirements in ASME Section III (Reference 6) described in Section 3.9. For qualification of non-safety-related, seismic Category I mechanical equipment is designed to provide reasonable assurance of structural integrity and its design intended function. For seismic qualification of active mechanical equipment, the methods and guidance in ASME QME-1-2007 (Reference 7), including Appendix QR-A, with exceptions provided in NRC RG 1.100, are used.

For procurement of equipment, the dynamic requirements for the seismic qualification are specified in the equipment’s design specifications. The equipment supplier is to submit a seismic qualification plan/procedure for review and approval prior to performing the seismic qualification. When test is employed, the equipment supplier is to submit a detailed test plan prior to conducting the test. When analysis is employed, the equipment supplier is to submit a detailed analysis procedure showing the methodology, approval, and description of the computer program used. If the plan/procedure is not acceptable, the seismic test plan or analysis procedure will be modified accordingly. The choice between testing and analysis may be made by the equipment supplier. However, the selected qualification program shall satisfy the requirements of the purchase specifications in accordance with the guidelines provided in IEEE Std. 344.

An existing seismic qualification is acceptable if it is properly documented, and if it meets all the requirements of the purchase specifications. The equipment supplier is to submit the seismic qualification documentation for review and approval prior to installation in the plant. The seismic qualification documentation is to include all the information stated in Subsection 3.10.4, to demonstrate that the equipment is qualified in accordance with the requirements of the purchase specifications.

3.10.1.2 Input Motion

The postulated dynamic loads related to the qualification of seismic Category I equipment are seismic loads (OBE and SSE), if applicable, hydrodynamic loads, and non-seismic loads.
loads (loads induced by pump trip, safety-relief valve open-case, etc.). The applicable loads are combined as part of the qualification of seismic Category I equipment.

These postulated dynamic loads are generally defined by the required response spectra (RRS). For in-line mounted equipment, they are defined by seismic coefficients. The location(s) in the plant will determine which spectra to use. Floor response spectra (FRS) are generated for specific buildings and elevations (floors) within a building as described in Subsection 3.7.2.5. When equipment is not directly mounted on floors, RRS reflects the amplification of the FRS due to the flexibility of equipment supporting structure. Selection of damping values for equipment to be qualified is made in accordance with NRC RG 1.61 (Reference 8) and IEEE Std. 344. Higher damping values are used only if justified by documented test data with proper identification of the source and mechanism. Margins are added to RRS for testing. Subsection 6.3.2.5 of IEEE Std. 323 recommends a 10 percent margin.

In considering the high-frequency seismic effect, the COL applicant is to investigate if site-specific spectra generated for the COLA exceed the APR1400 design spectra (CSDRS) in the high-frequency range. Accordingly, the COL applicant is to provide reasonable assurance of the functional performance of vibration-sensitive components in the high-frequency range (COL 3.10(2)).

The seismic qualification test/analysis will be performed for the components to envelop the in-structure response spectra resulting from the entire set of CSDRS, including ground motions for the COL sites with high frequency content. Evaluation of seismic Category I equipment for high frequency seismic input is addressed in Subsection 3.7B.7.4 and Technical Report, APR1400-E-S-NR-14004-P, Section 6.4 (Reference 13).

3.10.1.3 Selection of Qualification Method

The dynamic qualification of equipment is performed by analysis, testing, or a combination of testing and analysis. The dynamic qualification of equipment is concerned with the following.

a. Identifying which equipment must be qualified

b. Identifying what the safety-related function(s) or designed intended function(s) required of each piece of equipment is (are)
c. Defining the dynamic loads to be considered

d. Demonstrating the capability of the equipment to perform its function

e. Documenting that the process has been done in accordance with accepted regulatory and industry standards

In general, seismic Category I electrical equipment, for which functional operability must be demonstrated, is qualified by tests. Analysis alone is used for qualification of seismic Category I mechanical equipment if structural integrity alone can provide reasonable assurance of the design intended function.

For equipment whose functional operability cannot be demonstrated by analysis or testing because of its size, complexity, or the large number of similar configurations, a combination of test and analysis may be used. With the elimination of OBE, analysis checks for fatigue effects can be performed at a fraction of the SSE (Such as five one-half SSE events followed by one full SSE event or a number of fractional peak cycles equivalent to the maximum peak cycle for five one-half SSE events followed by one full SSE event).

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

Qualification of seismic Category I equipment and its supports meets the requirements of NRC RG 1.100 and IEEE Std. 344. Qualification methods of testing and analysis for confirming the functionality of equipment during and after an SSE, and for all static and dynamic loads from normal, transient, and accident conditions, are presented in this section.

3.10.2.1 Qualification by Analysis

The seismic analysis methods are in accordance with the guidance of IEEE Std. 344. Analysis without testing may be acceptable only if structural integrity alone can provide reasonable assurance of the design-intended function.

Procedures are presented that can be used to seismically qualify equipment by analysis for a number of OBEs followed by an SSE. Two approaches to seismic analysis are described. One approach is based on dynamic analysis, the other on static coefficient analysis.

a. Static coefficient analysis
This is an alternate method of analysis that allows a simpler technique in return for added conservatism. A determination of natural frequencies is not required. The acceleration response of the equipment is assumed to be the maximum acceleration in the amplified region peak of the RRS at a conservative and justifiable value of damping. A static coefficient of 1.5 has been established from experience to take into account the effects of multifrequency excitation and multimode response for linear frame-type structures, such as members physically similar to beams and columns, which can be represented by a simple model. A lower static coefficient may be used when it can be shown to yield conservative results. In a static coefficient analysis, the seismic forces on each component of the equipment are obtained by multiplying the values of the mass times the maximum peak of the RRS times the static coefficient. The resulting force should be distributed over the component in a manner proportional to its mass distribution. The stress at any point in the equipment can then be determined by combining the stress at that point due to the earthquake loading in each direction using the SRSS method.

b. Detailed dynamic analysis

Detailed dynamic analysis is basically finite element analysis. As such, it is generally used on equipment for which the simple models used in static-of-seismic-coefficient analysis are inadequate and/or on equipment for which a significant multimode response (when more than one resonant frequency is being excited simultaneously) or cross-coupling (when input in one direction results in response in one or more other directions) is anticipated.

c. Nonlinear equipment response

Nonlinearities may exist in addition to those associated with damping. These effects may be of a geometric nature, such as the closing of gaps, working of connections and rattling of components, or of a material source such as localized yielding. These effects may result in changing stiffness with increasing load. As frequency is also a function of stiffness, the frequencies may also change under increasing load. If a system exhibits significant nonlinearity, such behavior must be recognized and accounted for in any subsequent analysis so as to accurately predict the system response. If the nonlinearities cannot be adequately modeled, an alternative qualification method should be considered.
Nonlinearity may also occur as a result of local vibrations of equipment structure. One example is the high-frequency rattling of electrical cabinet doors that are not solidly secured in place. When such a condition exists and the operability of the mounted devices is deemed sensitive to this type of equipment nonlinear behavior, the analytic procedure must account for the behavior and must be properly validated.

d. Other dynamic loads

The analytical methodology described in this section for seismic loading is equally applicable to other dynamic loadings, such as hydrodynamic loadings.

e. OBE and SSE analysis

The analysis must show that OBE events followed by an SSE will not result in failure of the equipment to perform its safety function. When the OBE is defined as less than or equal to 1/3 of the SSE, explicit design or analysis is not required for the OBE. With the elimination of the OBE, the guidance for determination of the earthquake cycles described in SECY-93-087 (Reference 9) is used. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Annex D of IEEE Std. 344.

For floor-level excitation, this should be approximated by demonstrating that each excitation waveform will produce a response that includes the equivalent of at least 10 maximum peak-stress cycles. The number of OBEs and the fatigue-inducing potential per OBE are important only for low-cycle, fatigue-sensitive equipment. The analysis should determine that the structural integrity of the equipment is maintained in combination with other applicable loads during the OBE.

An analysis is performed using one of the previously described methods with five OBE events. Through the dynamic analysis or test, the equipment can be identified either rigid or flexible. For the determination of this characteristic, the modal analysis for the modeling of the equipment and any structural supports to express mass distribution and stiffness characteristics of the equipment and supports is performed to determine whether the equipment is rigid or flexible.
Rigid equipment can be analyzed using static analysis and seismic acceleration associated with the mounting location. Flexible equipment can be analyzed using the static coefficient method or using its dynamic response computed from a response spectrum, time history, or other analysis methods. The analysis determines that the structural integrity of the equipment is maintained in combination with other applicable loads during the OBE.

3.10.2.2 Qualification by Test

The seismic testing methods are performed in accordance with the guidance of IEEE Std. 344. Seismic testing is the most effective way of demonstrating operability, especially for instrumentation and for electrical components. Seismic testing is thus used to qualify these items of equipment. Generally, the seismic qualification by testing is conducted for equipment that cannot be qualified with analysis alone or equipment having components that potentially cause any malfunctions related to their intended functions.

Seismic testing is performed by subjecting equipment to vibratory motion that conservatively simulates movement at the equipment mounting while the operating conditions are simulated, and monitoring the performance of these devices during the test.

These postulated dynamic loads are generally defined by response spectra. The location(s) in the plant will determine which spectra to use. Floor response spectra are generated for specific buildings and elevations (floors) within a building as described in Subsection 3.7.2.5.

Seismic ground motion occurs simultaneously in all directions in a random fashion. Single-axis, biaxial, and triaxial testing is allowed. The single-axis test should be done in a conservative manner to account for the absence of input motion in some of the orthogonal directions. Single-axis and biaxial tests should be applied in a number of directions relative to the equipment to account for potential failure modes. An additional factor to be considered is the dynamic nature of the equipment (flexible or rigid) and the degree of cross-coupling.

a. Single-axis tests can be performed successively in the three principal orthogonal axed of the equipment at these times:

1) when it can be proved that the coupling is zero (or very low) between the three principal axes of the equipment, taken in pairs; or
2) when the equipment, in a seismic event, is subjected to a single excitation that may be considered as single-axis because of equipment mounting conditions. For example, if a device is normally mounted on equipment that amplifies motion in one direction, or if the method of mounting a device constrains its motion in a particular direction, single-axis testing may be adequate.

b. Biaxial testing is acceptable if the tests conservatively reflect the seismic event at the equipment mounting locations. Coupling in equipment is considered. Biaxial testing includes simultaneous input in horizontal and vertical axes. The selection of the horizontal axis may include the principal axes or some other direction selected to expose potential failure modes by testing the equipment in its most vulnerable direction.

Independent random inputs are preferred, and the test should be performed in two steps, with the equipment rotated 90 degrees in the horizontal plane for the second step. Statistical independence of the input motions should be reasonably assured.

If independent inputs are not used (i.e., pseudo-biaxial), four tests should be performed: Test 1 with the inputs in phase, Test 2 with one input 180 degrees out of phase, Test 3 with the equipment rotated 90 degrees horizontally and the inputs in phase, and Test 4 with the same equipment orientation as the third test but with one input 180 degrees out of phase.

c. Triaxial tests are desirable when significant couplings exist simultaneously between the three preferred axes of the equipment. Triaxial tests must be performed with a simulator capable of independent motions in all three orthogonal directions. The input motions should be statistically independent.

d. The test response spectra (TRS) should envelop the RRS over the frequency range of interest. The TRS should be computed with a damping value equal to or greater than that of the RRS. The shake table maximum peak acceleration should equal or exceed the ZPA of the RRS. The total test duration and number of equivalent maximum peak cycles should be per IEEE Std. 344.

Testing is performed to provide reasonable assurance that equipment can withstand the effects of seismic events and accidents and still be capable of performing its safety-related functions.
Seismic ground motion occurs simultaneously in all directions in a random fashion. Currently, single-axis, biaxial, and triaxial testing are allowed. A 10 percent margin is added on RRS during testing in accordance with Subsection 6.3.1.6 of IEEE Std. 323. The TRS must envelop the RRS in order for an item of equipment to be qualified (or justified). Sometimes, in a low-frequency area (below 3 Hz), TRS does not envelop RRS because of machine limitations. This requires justification based on the dynamic characteristics of the equipment. The TRS and RRS should be compared at the same damping value. A conservative TRS is greater than the RRS. Justification must be made when the TRS is less than the RRS.

Vibration aging testing may be performed preceding the OBE and SSE tests to show that the lower levels of normal and transient vibration associated with the plant operation will not adversely affect the equipment’s function.

Seismic qualification tests include five OBE test preceding the SSE for specified seismic events.

3.10.2.3 Operability of Active Equipment

The supplier is to prove the operability of all active equipment before, during, and after design basis events including seismic by test and/or analysis and provide the test or analysis report as a part of the dynamic qualification report.

3.10.2.3.1 Mechanical Equipment

The methods and procedures used for qualifying active mechanical equipment (i.e., valves and pumps) are described in Subsections 3.9.3, 3.10.2, and this subsection. Analysis, test, or a combination of test and analysis are used for qualification of seismic Category I active mechanical equipment to show it maintains structural integrity and functionality. The methods are used to provide reasonable assurance of equipment operability for its intended function under required plant conditions.

Seismic Category I active mechanical equipment is designed to withstand seismic and other dynamic loads, including the intended service load conditions in the equipment design specifications, in accordance with the requirements in ASME Section III described in Subsection 3.9.3.
Seismic qualification for active mechanical equipment is in accordance with IEEE Std. 344, ASME QME-1, and NRC RG 1.100 as stated in Subsection 3.10.2.

For mechanical equipment, the functionality by analysis and/or tests is proven as follows:

a. Pumps

A static deflection analysis and/or test for the shaft and rotor (if applicable) should be performed under design basis loading, including the maximum allowable nozzle loads specified in the equipment design specification. The deflection is less than the allowable/recommended deflection by the equipment supplier.

b. Valves

The following are the acceptable methods that can be applied to demonstrate valve operability:

1) Manual Valves

Active manual valves are those that should be opened or closed after DBA. In this case, the equipment supplier is to prove that the valve moving parts (stem, disc, etc.) are not permanently damaged due to DBA along with the maximum operating and nozzle loads by analysis and/or test.

2) Check Valves

The integrity of the valve and its parts, including disc, disc support, hinge, hinge-pin, hinge-arm, and seat is proven by test and/or analysis. The valve's operability verification document should address all possible worst loading conditions on the valve during and after seismic events.

3) Other Active Valves

For seismic qualification of all other active valves, the methods and guidance in ASME QME-1-2007 are used.

c. Mechanical Drive Turbines

The operability of the mechanical drive turbine focuses primarily on the operability of auxiliary active components (valves, pumps, instruments) associated
with or mounted on the turbine. The operability is determined by analysis and/or test.

d. Fans

Shaft and bearing deflections when the fan is subjected to the external design base loads are determined. The resulting clearance between the shaft and bearing as a result of these loads will be smaller than the recommended clearance by the manufacturer.

e. Diesel Engine

For the operability of the diesel engine and its auxiliary active components (valves, pumps, instruments), the methods described in NRC RG 1.9 (Reference 10) and IEEE Std. 387 (Reference 11) are used.

3.10.2.3.2 Electrical and Instrumentation

The supplier is to use the qualification test methods described in Subsection 3.10.2.2 to prove the operability of active electrical and instrumentation equipment.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation

Analyses or tests are performed for all supports of mechanical and electrical equipment to provide reasonable assurance of their structural capability.

The analytical results include the required input motions to the mounted equipment obtained in the manner stated in Subsection 3.10.1.2. Combined stresses of the designed component supports are maintained within the stress limits of the ANSI/AISC N690 including Supplement 2 (Reference 12). The loads, load combinations, combined stresses, and stress criteria to demonstrate the design adequacy of cable trays, conduits, and their supports are provided in Appendix 3.9A.

For supports of mechanical (ASME Section III) equipment, the analytical results include the loads, loading combinations, and combined stress limits described in Subsection 3.9.3 for ASME Section III, Class 1, 2, and 3 component supports. The jurisdictional boundary between ASME Section III, Class 1, 2, and 3 component supports and the building structure are established in accordance with ASME Section III, NF.
Supports are tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response at the equipment mounting location is monitored and characterized in the manner as stated in Subsection 3.10.2.2. In such a case, equipment is tested separately for operability and the actual input motion to the equipment in this test is to be more conservative in amplitude and frequency content than the monitored response from the support test.

3.10.4 Test and Analysis Results and Experience Database

Complete and auditable records are maintained for the life of the plant at the plant administrative facilities. These records are updated and kept current as equipment is replaced, further tested, or otherwise further qualified.

The COL applicant is to develop the equipment seismic qualification files that summarize the component’s qualification, including a list of equipment classified as seismic Category I in Table 3.2-1 and seismic qualification summary data sheets (SQSDS) for each piece of seismic Category I equipment (COL 3.10(3)). The SQSDS include the following information:

a. Identification of equipment, including vendor, model number, and location within each building. Valves that are part of the RCPB are identified.

b. Physical description, including dimensions, weight, and field mounting condition

c. A description of the equipment’s function within the system

d. Identification of all design (functional) specifications and qualification reports, and their locations

e. Description of the required loads and their intensities for which the equipment is qualified

f. If qualified by test, identification of the test methods and procedures, important test parameters, and a summary of the test results that includes test response spectra (TRS) enveloping required response spectra (RRS)
g. If qualified by analysis, identification of the analysis methods and assumptions and comparisons between the calculated and allowable stresses and deflections for critical elements

h. The natural frequency (or frequencies) of the equipment

i. Identification of whether the equipment is affected by vibration fatigue cycle effects and a description of the methods and criteria used to qualify the equipment for such loading conditions

j. Indication whether the equipment has met the qualification requirements

k. A compilation of the required response spectra (or time history) and corresponding damping for each seismic and dynamic load specified for the equipment together with all other loads considered in the qualification and the method of combining all loads

3.10.4.1 Implementation Program and Milestones

The COL applicant is to perform equipment seismic qualification for seismic Category I equipment and provide milestones and completion dates of equipment seismic qualification program (COL 3.10(4)).

3.10.4.2 Experience-based Qualification

Experience-based qualification is not used for any equipment.

3.10.5 Combined License Information

COL 3.10(1) The COL applicant is to provide documentation that the designs of seismic Category I SSCs are analyzed for OBE, if OBE is higher than 1/3 SSE.

COL 3.10(2) The COL applicant is to investigate if site-specific spectra generated for the COLA exceed the APR1400 design spectra in the high-frequency range. Accordingly, the COL applicant is to provide reasonable assurance of the functional performance of vibration-sensitive components in the high-frequency range.
COL 3.10(3) The COL applicant is to develop the equipment seismic qualification files that summarize the component’s qualification, including a list of equipment classified as seismic Category I in Table 3.2-1 and SQSDS for each piece of seismic Category I equipment.

COL 3.10(4) The COL applicant is to perform equipment seismic qualification for seismic Category I equipment and provide milestones and completion dates of equipment seismic qualification program.

3.10.6 References


3.11 Environmental Qualification of Mechanical and Electrical Equipment

This section presents the APR1400 approach for selecting and identifying equipment required to be designed and qualified in conformance with the requirements of 10 CFR Part 50, Appendix A, GDCs 1, 2, 4, and 23; 10 CFR Part 50, Appendix B, Quality Assurance Criteria III, XI and XVII, and 10 CFR 50.49.

Mechanical, electrical, and I&C equipment important to safety is qualified to meet its performance requirements under the environmental and operating conditions in which the equipment is required to function and during the length of time for which its function is required.

The APR1400 qualification approach conforms with the requirement of 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effect Design Bases,” and includes the scope of the following items:

a. Equipment associated with systems that are essential for emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise essential in preventing significant release of radioactive material to the environment

b. Equipment that initiates the above functions automatically

c. Equipment that is used by the operators to initiate the above functions manually

d. Equipment whose failure can prevent the satisfactory accomplishment of one or more of the safety functions specified in above item a.

e. Electrical equipment important to safety, as described in 10 CFR 50.49(b)(1), and (2) Certain post-accident monitoring equipment as described in 10 CFR 50.49(b)(3) and NRC RG 1.97.

f. Accident monitoring equipment specified in NRC RG 1.97

The implementation of the APR1400 Equipment Qualification is described in the KHNP Technical Report on the APR1400 Equipment Qualification Program (Reference 1), hereinafter referred to APR1400 EQP. Seismic qualification of the equipment is described in Section 3.10 and the APR1400 EQP in detail.
3.11.1 Equipment Location and Environmental Conditions

3.11.1.1 Equipment Location

Plant areas for equipment important to safety are divided into two based on the environmental conditions that potentially could occur within these areas as a result of a variety of plant events.

a. Harsh environment

An environment where a significant increase in pressure, temperature, relative humidity, or chemical environment occurs as a result of a design basis accident, or where a total integrated dose (TID) of greater than 100 Gy is predicted. In electronic components such as semiconductors, the total integrated dose is greater than 10 Gy. Detailed information is included in the APR1400 EQP.

b. Mild environment

An environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. Any area that is not a harsh environment is a mild environment.

Each zone is subdivided into various subzones according to the severity of environmental parameters and its detailed information on the identification and location of equipment and TID values for each room is described in Table 3.11-2 and APR1400 EQP, table 2, table 3.

The COL applicant is to identify and qualify the site-specific mechanical, electrical, I&C, and accident monitoring equipment specified in NRC RG 1.97 (COL 3.11(1)).

3.11.1.2 Definition of Environmental Conditions

The environmental conditions under which the equipment performs its design safety functions include all normal, anticipated operational occurrences (AOOs), accident, and post-accident conditions due to design basis accidents (DBAs), and are defined as follows:

a. Normal condition
The environmental conditions expected during normal plant operation with systems operating normally.

b. Anticipated operational occurrence condition

The limiting environmental conditions expected following an event or transient that is not a normal operating condition but is not considered as an accident.

c. Accident condition

The environmental conditions that would occur during a loss-of-coolant accident (LOCA), main steam line break (MSLB), and high-energy line break (HELB).

d. Post-accident condition

The environmental conditions expected following design basis accident.

The environmental parameters such as temperature, pressure, relative humidity, radiation, and chemical spray for various locations throughout the plant are provided in Figure 3.11-1 through 7, Table 3.11-2, and the APR1400 EQP.

3.11.1.3 Equipment Operability Times

The required operational time during which each equipment is required to operate in the accident environment is identified in Table 3.11-2, and defined as follows:

a. Continuous

Component is required to operate throughout the design basis accident without interruption (i.e., at least 6 months).

The following types of equipment are examples of this category:

- Electrical power source & distribution equipment: SWGR, MCCs, batteries, battery chargers, inverter, cables and other electrical equipment needed to provide electrical power to safety-related equipment.

- Monitoring equipment: Sensors, transmitters and monitoring panels
- Continuous motors: Continuous operation expected motor such as pump motor

b. Short-term

Component is required to operate one time up to 24 hours from the start of the design basis accident.

The following types of equipment are examples of this category:

- Isolation valves: valves’s safety function that is completed by one time operation

c. Intermittent

Component is capable of operating throughout the design basis accident (i.e., at least 6 months), starting and stopping on an as-needed basis.

The following types of equipment are examples of this category:

- Valves: Modulating, those that require intermittent operation depending on system situation

d. Varies

Component is capable of operating throughout the design basis accident (at least 6 months) depending on the situation, but it is not needed if something else can perform the same task.

Any equipment with an operability time labeled as “varies” will be operational for the duration of the most limiting design basis accident.

3.11.2 Qualification Tests and Analyses

Environmental qualification of Class 1E equipment is in accordance with the requirements of 10 CFR 50.49, NRC RG 1.89 (Reference 2), and IEEE Std. 323 (Reference 3). Equipment qualification standards that are available are also met.
A description of the qualification method is contained in a qualification report for each type of equipment. The qualification method encompasses appropriate combinations of any or all of the type testing, operating experience, and analysis.

The typical approach used for the qualification of equipment potentially exposed to a harsh environment is as follows:

a. Evaluate the equipment life capability according to the planned design life service exposure to normal and abnormal environmental conditions plus the worst-case accident exposure condition.

b. Use the Arrhenius methodology on weak link materials if test data exist to derive the qualified life indicated above item.

Environmental qualification of mechanical equipment conforms with GDCs 1 and 4, and 10 CFR Part 50, Appendix B, Criteria III and XVII which requires:

a. Components shall be designed to be compatible with the postulated environmental conditions including those associated with LOCAs.

b. Qualification records shall be maintained and shall include the results of tests and material analyses.

c. Design control measures shall be established for verifying the adequacy of design

Mechanical equipment is principally divided into active and non-active (passive) mechanical equipment.

Environmental qualification of mechanical equipment is focused on the materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) and is limited to active mechanical equipment located in harsh environment which has mechanical moving parts to perform its safety-related function, and the qualification effort requires the evaluation of all safety-related nonmetallic parts against the applicable environmental conditions.

Non-active mechanical equipment (passive mechanical equipment) whose safety function is structural integrity is designed and qualified for the appropriate temperature and pressure environment in accordance with the applicable code to which it is constructed such as ASME Boiler and Pressure Vessel Codes.
The followings shall be confirmed in implementing the environmental qualification of active mechanical equipment:

a. To identify safety-related mechanical equipment located in harsh environment areas, including its required operating time

b. To identify nonmetallic subcomponents of such equipment

c. To Identify the environmental conditions and process parameters for which this equipment must be qualified

d. To identify nonmetallic material capabilities.

The COL applicant is to identify the nonmetallic parts of mechanical equipment in procurement process (COL 3.11(2)).

e. To evaluate environmental effects

The service requirements and the environmental requirements are defined in the APR1400 design specification for this equipment.

Materials are selected based on extensive testing and long-time service that is compatible with the requirements. Quality assurance of design and quality control of processes provide reasonable assurance that the component meets the specification requirements. Further, the design and manufacturing organizations certify compliance. In-service surveillance and maintenance programs, followed by refurbishment or replacement of parts if necessary, provide further assurance that the safety equipment will remain operable.

The evaluation of environmental adequacy of the equipment is initiated by the full definition of environmental requirements in APR1400 design specification, as stated above. Test reports and analyses that substantiate operability after exposure to the environment, and the quality assurance documentation, are to be filed by the operating licensee.

The results of environmental qualifications are included as a part of the equipment qualification reports. These reports are used to establish the maintenance and repair plan for the equipment and procurement of the parts for the life of the plant. If the components are replaced or qualified by other methods, the reports should be traceable.
The safety-related active mechanical equipment that contains non-metallic parts is also environmentally qualified in accordance with ASME QME-1-2007, “Qualification of Active Mechanical Equipment,” QR-B, “Guide for Qualification of Non-Metallic Parts” (Reference 9.31), as endorsed by NRC RG 1.100, and is listed in Table 3.11-3 as specified by NRC RG 1.206, C.I.3.11.6.

3.11.2.1 Environmental Qualification during Normal Operation

Equipment which is not significantly affected environmentally by the design basis accident (DBA) is said to exist in a mild (normal plus abnormal service conditions) environment. For the qualification of both electrical and mechanical equipment in a mild environment, a qualified life is not required if no significant aging mechanism in mild conditions is identified in accordance with IEEE Std. 323. If the predicted life based on experience, aging analysis, or tests is less than the design life of the plant, that equipment is subjected to a surveillance program and a preventative maintenance program that restores it to qualified operability. The detailed maintenance or surveillance program for specific plants is to be developed based on the specific equipment for the APR1400 and the results of qualification testing and analysis for that equipment.

The ranges of the design temperatures, pressures, relative humidity, and radiation for typical mild environment areas in which safety-related equipment is located are provided in Table 3.11-2.

3.11.2.2 Environmental Qualification during and after a Design Basis Accident

Equipment located in harsh environments is designed to remain functional in the environment that exists at the equipment location, for the length of time during and after the DBA for which it is required to be functional, and for the integrated radiation dose during normal operation. Replacement of equipment will be made before the maximum calculated DBA conditions that could result in exceeding design limits for a piece of equipment, when considering the cumulative effects of the normal operating conditions that each piece of equipment has received, as well as the maximum cumulative worst case conditions calculated for the worst case accident. The temperature, pressure, and humidity environment inside the containment after a LOCA and MSLB is discussed in detail in Subsections 6.2.1.3 and 6.2.1.4. The containment spray characteristics are given in
Subsection 6.2.2.1. The worst-case integrated post-accident radiation doses for those areas at which equipment is located are provided in Table 3.11-2.

10 CFR Part 50, Appendix A, GDCs 1, 4, 23, and 50 are met as discussed in Subsections 3.1.1, 3.1.4, 3.1.19, 3.1.43, and 6.2.1, respectively.

The requirements of 10 CFR Part 50, Appendix B, Quality Assurance Criterion III, are met as discussed in Section 17.5 and applicable Regulatory Guides and codes include the following:

a. Regulatory Guide 1.40, Qualification of Continuous-Duty Safety-Related Motors for Nuclear Power Plants

b. Regulatory Guide 1.63, Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants

c. Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

d. Regulatory Guide 1.89 Environmental Qualification of Certain Electric Equipment Important to Safety in Nuclear Power Plants

e. Regulatory Guide 1.97, Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants

f. Regulatory Guide 1.151, Instrument Sensing Lines

g. Regulatory Guide 1.156, Environmental Qualifications of Connection Assemblies for Nuclear Power Plants

h. Regulatory Guide 1.158, Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants


The COL applicant is to address operational aspects for maintaining the environmental qualification status of components after initial qualification (COL 3.11(3)).

The COL applicant is to provide a full description of the environmental qualification of mechanical and electrical equipment program (COL 3.11(4)).

The operational aspects to address development of the operational program to maintain the environmental qualification status of components, consistent with SRP 3.11 will include the following:

a. evaluation of environmental qualification results throughout design life to establish activities that support continued environmental qualification

b. determination of surveillance and preventive maintenance activities based on environmental qualification results and operating experience

c. consideration of environmental qualification maintenance recommendations from equipment vendors

d. evaluation of operating experience in updating surveillance and preventive maintenance activities for specific equipment
e. development of plant procedures that specify individual equipment identification, appropriate references, installation requirements, surveillance and maintenance requirements, post-maintenance testing requirements, condition monitoring requirements, replacement part identification, and applicable design changes and modifications

f. development of plant procedures for reviewing equipment performance and environmental qualification operational activities, trending the results, and incorporating lessons learned through appropriate modifications to the environmental qualification operational program

g. development of plant procedures for the control and maintenance of environmental qualification records

Safety-related passive pressure boundary components are designed and qualified for the appropriate temperature and pressure environment in accordance with the requirement of ASME Boiler and Pressure Vessel Code, Section III. The qualification of non-metallic parts in safety-related passive mechanical equipment is ensured through the means specified in Subsection 3.11.2.3.

The materials used in the fabrication of mechanical and structural components inside the containment are selected to minimize corrosion and hydrogen generation resulting from contact with spray solutions. The use of aluminum and zinc is minimized in these components.

Material incompatibilities at interfaces are demonstrated under the worst case environmental conditions that it will be exposed to.

3.11.2.3 Environmental Qualification Method

a. Qualification by test

Qualification testing is performed on actual equipment to stimulate normal, abnormal, and accident conditions. While testing, the specimen is subjected to accelerated aging. Synergistic effects are considered in the aging program where synergistic effects have been identified on materials that are included in the equipment being qualified. When size or other practical requirements limit or preclude the type testing, this part of demonstration is completed by use of
operating experience, analysis of partial type test data, or combinations of these qualification.

b. Qualification by analysis

If qualification documentation for other equipment is available, it is reviewed to determine if the qualified equipment is similar to that being procured. If the former is enveloped by the latter, then an analysis to determine qualification life is performed using the existing data.

In addition, if extrapolation and interpolation techniques to extend the application of test data (basically, equipment similarity analysis) are used, the following criteria should be met:

Material:

Materials of construction shall either be the same or equivalent. Any identified differences shall be shown not to adversely affect performance of the safety function(s).

Size:

Size may vary if the basic configuration remains the same and dimensions are related to known scaling factors. Consideration shall be taken of such factors as thermal effects of different surface areas and seismic effects of different masses and modes.

Shape:

The shape shall be the same or similar (subject to restrictions of size) and any differences shown shall not adversely affect the performance of the safety function(s).

Stress:

Operating and environmental stresses on the new equipment shall be equal to or less than those experienced on the qualified equipment under normal and abnormal conditions.
Aging Mechanisms:

The aging mechanisms that apply to the tested equipment encompass those that apply to the similar equipment.

Function:

The safety function(s) as evaluated shall be the same.

c. Qualification by operating experience

Qualification of equipment using operating experience is used in combination with supporting documentation as a basis for environmental qualification if certification of conformance by the vendor is not feasible. This type of qualification may be used for equipment for which testing is not feasible due to the equipment physical size. This evaluation is performed using similar equipment with a successful operating history in a service environment equal to or more severe than the environment for the equipment in question. If similarity analysis is used, the extrapolation and interpolation techniques mentioned in Subsection 3.11.2.3.b shall be used. The validity of operating experience as a means of qualification is determined from the type and amount of available supporting documentation, the service conditions, and equipment performance.

d. Combined qualification

Combined qualification is used for any equipment that cannot be qualified through a full type test. Combined qualification usually entails type test, previous operating experience, and analysis. Partial type tests with extrapolation or analysis, operating experience with extrapolation or analysis, and type tests supplemented with tests of components and analysis are examples of combined qualification. The qualification program for the emergency diesel generator utilizes a combined qualification technique.

Aging for Harsh Environment Equipment

Equipment that is located in zones susceptible to a harsh environment is also exposed to a mild environment before DBA. Such equipment undergoes an aging analysis that focuses on the identification of aging mechanisms that significantly increase the equipment's susceptibility to DBA. If no known significant aging mechanisms are found, a
surveillance or preventive maintenance program is developed to monitor for degradation. If an aging mechanism is found that is known to significantly degrade the equipment, that mechanism is analyzed to determine whether an accelerated aging program or a periodic part replacement program is appropriate.

**Radiation for Harsh and Non-Harsh Environment Equipment**

Equipment is designed for the types and levels of radiation associated with its location. The design includes the normal operation contribution plus the radiation associated with the limiting DBA for which the equipment is required to be functional and for the duration of time both during and after the accident for which it is required to be operational. The levels that are defined in Table 3.11-2 are the worst-case values and are intended to represent an upper-bound dose value for that region.

Equipment that is exposed to radiation more than 100 Gy, exceeding 10 Gy for electronic equipment, is irradiated to its anticipated TID before type testing unless determined by analysis that radiation does not affect its ability to perform its required function. Where the application of the accident dose is planned during DBA testing, it need not be included during the aging process.

A TID of 100 Gy or less, not exceeding 10 Gy for electronic equipment, does not affect the strength or properties of material used. Therefore, further qualification analysis and tests for components that are exposed to lesser radiation are not necessary.

Mechanical and electrical equipment is qualified to appropriate radiation environments as discussed previously. If more than one type of radiation is significant, the radiation types such as gamma, beta, or neutron may be applied separately.

**Gamma**

Safety-related equipment is tested to gamma radiation levels developed. Upper-bound dose values for various plant regions are provided in Table 3 of APR1400-E-X-NR-14001-P.

**Beta**

Equipment exposed to beta radiation is identified, and an analysis is performed to determine whether the operability of the equipment is affected by beta radiation ionization and heating effects. Qualification is performed by test unless analysis demonstrates that the safety function is not degraded by beta exposure. Equipment is tested or analyzed to
the beta radiation levels defined in Table 3 of APR1400-E-X-NR-14001-P. Where testing is recommended, a gamma equivalent radiation source is used.

**Neutron**

Equipment exposed to neutron radiation is identified and neutron radiation levels defined. When actual neutron dose qualification testing is not performed, an equivalent gamma radiation dose is used for qualification testing to simulate neutron exposure. The basis for establishing an equivalent gamma radiation dose is provided in Table 3 of APR1400-E-X-NR-14001-P.

**Chemical spray**

After a postulated accident, such as a LOCA or MSLB, components located in the reactor containment building are exposed to a chemical spray. Equipment is environmentally tested to these conditions, and performance requirements are demonstrated during and after the test. The most severe spray composition is determined by single failure analysis of the spray system. Corrosion effects due to long-term exposure are addressed, as appropriate.

Where qualification for a chemical spray environment is required, the simulated spray is initiated.

Typical values of chemical spray composition, concentration and pH are defined in Table 4 of APR1400-E-X-NR-14001-P.

**Humidity**

Equipment that is adversely affected by a high-humidity environment and required to operate in a high-humidity environment but not subjected to a steam environment during DBA testing is environmentally qualified by type test. The equipment is performance tested before the application of the high-humidity environment to establish a baseline, then retested while exposed to a humid environment that envelops the required humidity condition, and again retested after removal of the high-humidity environment for comparison to the original baseline measurement. Comparison of the baseline tests determines if any degradation is present and ensures that operability criteria are met. Equipment that is subjected to steam environments is subjected to the appropriate test profiles.
Submergence

Equipment locations and operability requirements are reviewed to establish whether or not specific equipment could be subject to submergence during its required operating time.

Flood levels both inside and outside containment are reviewed and potential impacts on equipment qualification are appropriately addressed. Equipment that is required to operate underwater is qualified by type test.

Power Supply Voltage and Frequency Variation

Power supply voltage and frequency variation is addressed in the equipment design and verification process. During the design process, the range of power supply variation is determined. Equipment specifications incorporate the ranges to ensure acceptable operation. Type testing of the equipment at the extremes of power supply variations is performed if required.

Synergistic effects

Synergistic effects are evaluated to verify that these effects do not adversely affect the qualification of the mechanical, electrical, and I&C equipment, as required in accordance with 10 CFR 50.49(e)(7).

Electromagnetic interference and radio-frequency interference

Testing for electromagnetic interference and radio-frequency interference (EMI/RFI) and power surges is performed with the guidance of NRC RG 1.180.

3.11.3 Qualification Test Results

The COL applicant is to document the qualification test results and qualification status in an auditable file for each type of equipment in accordance with the requirements 10 CFR 50.49(j) (COL 3.11(5)). Because EQP is an operational program, the COL applicant is to describe the EQP implementation milestones based on the APR1400 EQP (COL 3.11(6)).

3.11.3.1 Electrical and I&C Equipment

A list of the instrumentation and electrical equipment required to achieve safe shutdown or mitigate the effect of a DBA is presented in Table 3.11-3. Required equipment that is
potentially exposed to harsh environments during an event is environmentally qualified. This includes the equipment that acts to safely shut down the reactor, provides adequate core cooling, isolates the containment, provides residual heat removal, and precludes uncontrolled release of radioactive effluents. The results of qualification testing and analysis for Class 1E equipment are described in their qualification reports.

3.11.3.2 Mechanical Equipment

Mechanical equipment is relatively insensitive to environmental conditions considering that service conditions usually far exceed environmental conditions. For mechanical equipment, the service requirements and the environmental requirements are defined in the design specification. Materials are selected based on extensive testing and long service, which is compatible with the requirements. Quality assurance of design and quality control of processes provides reasonable assurance that the component meets the specification requirements. Further, the design and manufacturing organizations certify conformance. In-service surveillance and maintenance programs, followed by refurbishment or replacement of parts if necessary, provide further assurance that the safety equipment remains operable.

The evaluation of environmental adequacy of equipment is initiated by the full definition of environmental requirements in equipment specifications, as stated above. Test reports and analyses that substantiate operability after exposure to the environment, and the quality assurance documentation, are to be filed by an operator.

The results of environmental qualifications are included as a part of the equipment qualification reports. These reports are used to establish the maintenance and repair plan for the equipment and procurement of the parts for the life of the plant. If the components are replaced or qualified by other methods, the reports should be traceable.

The safety-related mechanical equipment that contains non-metallic parts is environmentally qualified in accordance with QR-B, “Guide for Qualification of Non-Metallic Parts” of ASME QME-1-2007 “Qualification of Active Mechanical Equipment,” (Reference 9.31), as endorsed by NRC RG 1.100, and is listed in Table 3.11-2 as specified by NRC RG 1.206, C.I.3.11.6.
3.11.4 Loss of Ventilation

The need for the heating, ventilation, and air conditioning (HVAC) systems and the design bases that prevent the loss of safety-related ventilation are described in Sections 6.4 and 9.4. The two-division concept provides 100 percent redundancy of all safety-related equipment and the HVAC system. In event of a failure of one system to deliver the desired conditioned air, the second system is energized automatically in its place. This changeover is also designed to be achieved manually.

All of the areas that are designed with HVAC listed in Table 3.11-1 are cooled with chilled water, which provides reasonable assurance of a temperature level below that for which the equipment is qualified. Therefore, temperature switches are provided in each room.

The vital instrument and equipment in the main control room (MCR), Class 1E switchgear room, battery room, and remote shutdown console room are served with 100 percent redundancy of the HVAC unit.

The containment has standby cooling and ventilation equipment for each of the component parts of the ventilation system to maintain normal equipment qualification conditions. The containment ventilation systems are not credited in maintaining post-accident environmental conditions.

The chilled water system is divided into two circuits, non-essential and essential, in each division. One essential chiller and one essential chilled water pump serve the safety-related cubicle coolers in each division. The chilled water temperature in the essential circuit ranges from 5.6 °C to 11.1 °C (42 °F to 52 °F), which provides humidity and temperature control.

Class 1E equipment that is located in the MCR or similar areas includes the following:

a. Plant protection system (PPS)

b. Main control panels

c. Auxiliary process cabinet (APC)

Other instrumentation, such as process transmitters and signal converters and the reactor trip switchgear system circuit breakers, is located in the auxiliary building or containment building. Equipment in these areas is qualified for the maximum expected temperature,
radiation, humidity, and pressure under which and the duration both during and following
the accident for which the equipment is expected to be functional.

3.11.5 Estimate Chemical and Radiation Environment

3.11.5.1 Chemical Environment

After a postulated accident, such as the LOCA or MSLB, components located in the reactor
containment building are exposed to chemical spray. Equipment is environmentally tested
to these conditions, and performance requirements are demonstrated during and after the
test. The most severe spray composition is determined by single failure analysis of the
spray system. Corrosion effects due to long-term exposure are addressed, as appropriate.
The components of engineered safety features (ESFs) inside the containment are designed
to perform their safety-related functions in a chemical environment resulting from the boric
acid recirculated through the safety injection system (SIS) and containment spray system
(CSS). The SIS and CSS are designed to perform their functions under the conditions of
the maximum and long term boric acid concentration and pH.

3.11.5.2 Radiation Environment

Safety-related components are designed to ensure acceptable performance, taking into
consideration normal operational radiation exposure in addition to the single most adverse
post-accident environment for which they are required to be functional.

The radiation qualifications for individual safety-related components are developed
based on:

a. the radiation environment expected at the component location from equipment
   installation up to the time the equipment is required to remain functional
   postaccident, and

b. the limiting DBA for which the component provides a safety function.

The components in the ESF and the reactor protection systems are designed to provide
reasonable assurance of acceptable performance under normal operational radiation
exposure in addition to the single most adverse post-accident environment. The normal
operational exposures are based on the design source terms presented in Table 11.1-2,
which are the design basis source terms based upon 1.0% failed fuel with continuous gas
stripping operation.
In accordance with NRC RG 1.89, the source terms based on 1 percent fuel defect are used to calculate the TID during normal operation. The TID is calculated at a distance of 30.48 cm (1 ft) away from the equipment surface, and 60 years of continuous operation at full power is assumed.

For the equipment located in fuel transfer tube and cask loading pit that is used only during the refueling operation, the TID is calculated assuming that the radiation sources affect the equipment only during the refueling period. The TID during normal operation in these areas is negligible since no radiation source exists. Since these equipment do not perform safety-related functions, dose contributions from accident conditions are not considered. A 1-month duration refueling period is assumed for every 18 months of normal operation. Therefore, the TID is calculated based on 40 months of refueling operation during the plant life of 60 years. An additional safety margin of 20 percent is applied to the normal TID considering the potential contribution of radiation from adjacent cubicles. This margin bounds the dose contributions from adjacent sources for most of the cubicles. In cases where the sum of the dose contributions is greater than the margin, the actual dose rates are taken into account in determining the normal TID of the corresponding area.

Accidental TIDs are determined based on the limiting design basis accident to create the expected worst-case environmental conditions (i.e., bounding) for the structure, system, and components. Radiation environments for the components for which the most adverse accident conditions are post-LOCA, are based on the source term assumptions consistent with NRC RG 1.183. Radiation environments for the components for which the most adverse accident conditions are other than the LOCA, such as main steam line break, feedwater line break, or control element assembly (CEA) ejection, are based on conservative estimates of the fuel assembly gap activities and maximum reactor coolant specific activities as discussed in Section 11.1.

Post-accident ESF system and component radiation exposures are dependent on equipment location. In the containment and control room area, exposures are based on a postulated design basis LOCA. Source terms and other accident parameters are presented in Subsection 12.2.2 and Chapter 15 and are consistent with the recommendations of NRC RG 1.183.

In the auxiliary building, exposures are based on the assumption that significant portion of the core fission product inventory are recirculated in the containment sump water plus other post-accident airborne radioactivities as presented in Table 12.2-26. In the fuel handling
area, exposures are based on a fuel handling accident. Source terms and other accident parameters are presented in Chapter 15.

Organic materials that are within the containment are identified in Subsection 6.1.2. The design radiation exposures are based on gamma and beta radiation.

Equipment is designed for the types and levels of radiation associated with its location. The design includes the normal operation contribution plus the radiation associated with the limiting DBA for which the equipment is required to be functional and for the duration of time both during and after the accident for which it is required to be operational.

Equipment that is exposed to radiation of more than 100 Gy, exceeding 10 Gy for electronic equipment, is irradiated to its anticipated TID prior to type testing unless determined by analysis that the radiation does not affect its ability to perform its required function. Where the application of the accident dose is planned during DBA testing, it is not included during the aging process. A total integrated dose of 100 Gy or less, not exceeding 10 Gy for electronic equipment, does not affect the strength or properties of material used. Therefore, further qualification analysis and tests for components that are exposed to lesser radiation are not necessary.

Mechanical and electrical equipment are qualified to appropriate radiation environments. If more than one type of radiation is significant, the radiation types such as gamma, beta, or neutron may be applied separately.

With respect to the thermal conductivity degradation (TCD) effect of fuel pellets on the EQ of safety-related equipment in containment, the increased containment pressure and temperature are estimated to be negligible. The detailed description of the TCD effects on the containment integrity is provided in Reference 16.

3.11.6 Qualification of Mechanical Equipment

As mentioned in Subsection 3.11.3.2, for mechanical equipment located in a harsh environment, conformance with the environmental design provisions of GDC 4 is generally achieved by demonstrating that the nonmetallic parts/components (e.g., seal, gasket, lubricants, etc.) are suitable for the postulated design basis environmental conditions.

The effect of process medium temperature and radiation on the nonmetallic parts is evaluated for any process medium whose temperature and radiation are higher than the
highest external environmental temperature and radiation, and the combined effect of time-
temperature and radiation degradation is considered. Consideration is also given to process pressure, process media type and chemistry, and process humidity.

Nonmetallic parts mainly consist of seals, gaskets, and lubricants whose failure of leakage, interception or wear could lead to hindrance of the safety function in the equipment in which they are installed. The safety-related mechanical equipment that contains non-metallic parts is environmentally qualified in accordance with QR-B, “Guide for Qualification of Non-Metallic Parts” of ASME QME-1-2007 “Qualification of Active Mechanical Equipment,” (Reference 9.31), as endorsed by NRC RG 1.100, and is listed in Table 3.11-2 as specified by NRC RG 1.206, C.I.3.11.6.

3.11.7 Combined License Information

COL 3.11(1) The COL applicant is to identify and qualify the site-specific mechanical, electrical, I&C, and accident monitoring equipment specified in NRC RG 1.97.

COL 3.11(2) The COL applicant is to identify the nonmetallic parts of mechanical equipment in procurement process.

COL 3.11(3) The COL applicant is to address operational aspects for maintaining the environmental qualification status of components after initial qualification.

COL 3.11(4) The COL applicant is to provide a full description of the environmental qualification of mechanical and electrical equipment program.

COL 3.11(5) The COL applicant is to document the qualification test results and qualification status in an auditable file for each type of equipment in accordance with the requirements 10 CFR 50.49(j).

COL 3.11(6) The COL applicant is to describe the EQP implementation milestones based on the APR1400 EQP.

COL 3.11(7) The COL applicant is to provide the room number designation for equipment with unidentified rooms in Table 3.11-2, and ensure that the equipment is located within the environmental and radiation conditions specified.
3.11.8 References


Table 3.11-1

Ventilation Areas

<table>
<thead>
<tr>
<th>Area</th>
<th>Safety-Related</th>
<th>Area Temp. Alarm in Control Room</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Yes</td>
<td>No</td>
<td></td>
</tr>
<tr>
<td>Control Room Area</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Control Room and Adjacent Offices</td>
<td>O</td>
<td>O</td>
<td>O</td>
</tr>
<tr>
<td>2. Computer Room</td>
<td>O</td>
<td>O</td>
<td>O</td>
</tr>
<tr>
<td>Auxiliary Building Clean Area</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Turbine Driven AFP Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>2. Motor Driven AFP Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>3. Essential Chiller and Pump</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>Auxiliary Building Controlled Area</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. SC Heat Exchanger Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>2. SI Pump Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>3. CS Pump Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>4. CS Heat Exchanger Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>5. Penetration Room</td>
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<td>O</td>
<td></td>
</tr>
<tr>
<td>6. CCW Pump Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>Electrical Equipment Room</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Vital Instrument &amp; Equipment</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>2. Class 1E Switchgear Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>3. Class 1E Battery Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>4. Remote Shutdown Console Room</td>
<td>O</td>
<td>O</td>
<td></td>
</tr>
<tr>
<td>Fuel Handling Area</td>
<td></td>
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<tr>
<td>1. Spent Fuel Pool Heat Exchanger Room and Pump Room</td>
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<tr>
<td>2. Emergency ACU Room</td>
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<tr>
<td>Emergency Diesel Generator Area</td>
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<tr>
<td>Containment Building</td>
<td>O</td>
<td>O</td>
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### Auxiliary Feedwater System

<table>
<thead>
<tr>
<th>Tag No.</th>
<th>Equipment Identification</th>
<th>Equipment Function</th>
<th>Building</th>
<th>Room No.</th>
<th>Required Operational Time</th>
<th>Environmental Condition (1)</th>
<th>Radiation Condition (2)</th>
<th>Designation</th>
<th>Influence of Immersion (3)</th>
<th>Seismic Category</th>
<th>Remarks</th>
<th>Classification</th>
<th>HF Sensitive</th>
</tr>
</thead>
<tbody>
<tr>
<td>AFW-PPN01A</td>
<td>Turbine Driven Aux. Feedwater Pumps</td>
<td>ESF</td>
<td>AB</td>
<td>078-A13C</td>
<td>Continuous</td>
<td>Harsh</td>
<td>Mild</td>
<td>Mechanical EQ</td>
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<td>1 (3)</td>
<td>SR</td>
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<td></td>
</tr>
<tr>
<td>AFW-PPN01B</td>
<td>Turbine Driven Aux. Feedwater Pumps</td>
<td>ESF</td>
<td>AB</td>
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<td>Harsh</td>
<td>Mild</td>
<td>Mechanical EQ</td>
<td>No</td>
<td>1 (3)</td>
<td>SR</td>
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<td></td>
</tr>
<tr>
<td>AFW-PPN02A</td>
<td>Motor Driven Aux. Feedwater Pumps</td>
<td>ESF</td>
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<td>ESF</td>
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<td>Mild</td>
<td>Mechanical EQ</td>
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<tr>
<td>AFW-V0035</td>
<td>Globe Valve and Actuator, AFW Modulating</td>
<td>ESF</td>
<td>AB</td>
<td>COL.3.16(7)</td>
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<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>ESF</td>
<td>AB</td>
<td>COL.3.16(7)</td>
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<td>Harsh</td>
<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>AFW-V0037</td>
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<td>ESF</td>
<td>AB</td>
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<td>Harsh</td>
<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>AFW-V0043</td>
<td>Gate Valve and Actuator, AFW Isolation, CIV</td>
<td>ESF</td>
<td>AB</td>
<td>COL.3.16(7)</td>
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<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>Harsh</td>
<td>Mechanical EQ</td>
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### Auxiliary Feedwater Pump Turbine System

<table>
<thead>
<tr>
<th>Tag No.</th>
<th>Equipment Identification</th>
<th>Equipment Function</th>
<th>Building</th>
<th>Room No.</th>
<th>Required Operational Time</th>
<th>Environmental Condition (1)</th>
<th>Radiation Condition (2)</th>
<th>Designation</th>
<th>Influence of Immersion (3)</th>
<th>Seismic Category</th>
<th>Remarks</th>
<th>Classification</th>
<th>HF Sensitive</th>
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</thead>
<tbody>
<tr>
<td>AT-V0007</td>
<td>Globe Valve and Actuator, AFW Pump Turbine Steam Line Drain</td>
<td>ESF</td>
<td>AB</td>
<td>COL.3.16(7)</td>
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<td>Harsh</td>
<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>1 (3)</td>
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<td>AT-V0008</td>
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<td>Harsh</td>
<td>Mechanical EQ</td>
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<td>AT-V0009</td>
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<td>Harsh</td>
<td>Mechanical EQ</td>
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### Auxiliary Feedwater Storage and Transfer

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The table lists various equipment and their corresponding functions, operational times, environmental conditions, and designations, among other details. Each entry specifies whether the item is SR, SR, or O sensitive, and whether it is classified as SR or not.
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**Emergency Diesel Generator System**

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3.11-32

Rev. 3
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## APR1400 DCD TIER 2

### Emergency Diesel Engine Starting Air System

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### Emergency Diesel Engine Starting Air System

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- Environmental Condition (1): Mild
- Radiation Condition (2): Continuous, Short-Term
- Designation: Electrical EQ (EMC), Mechanical EQ
- Influence of Immersion (3): No
- Seismic Category: SR
- Remarks: (3)
- Classification: SR
- HF Sensitive: O

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3.11-35

Rev. 3
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**Main Steam System**

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**Primary Sampling System**

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**Reactor Coolant System**

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3. Designation: EQ/EMC
4. Influence of Immersion: No, I
5. Seismic Category: SR
6. Remarks
7. Classification: SR, O
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## Table 3.11-2

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**Shutdown Cooling System**

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**Essential Service Water System**

<p>| SX-P01A  | Essential Service Water Pump                                | ESF                | ESWPB    | COL.3.11(7) | Continuous               | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-P01B  | Essential Service Water Pump                                | ESF                | ESWPB    | COL.11(7)  | Continuous               | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-P02A  | Essential Service Water Pump                                | ESF                | ESWPB    | COL.11(7)  | Continuous               | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-P02B  | Essential Service Water Pump                                | ESF                | ESWPB    | COL.11(7)  | Continuous               | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-V045  | ESW Pump 01A Discharge                                      | ESF                | ESWPB    | COL.11(7)  | Varies                  | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-V046  | ESW Pump 01B Discharge                                      | ESF                | ESWPB    | COL.11(7)  | Varies                  | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-V047  | ESW Pump 02A Discharge                                      | ESF                | ESWPB    | COL.11(7)  | Varies                  | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-V048  | ESW Pump 02B Discharge                                      | ESF                | ESWPB    | COL.11(7)  | Varies                  | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     | (1) | SR         | O           |
| SX-V1001 | ESW Pump 01A Discharge Check                                | ESF                | ESWPB    | COL.11(7)  | Continuous               | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     |     |            |             |
| SX-V1002 | ESW Pump 01B Discharge Check                                | ESF                | ESWPB    | COL.11(7)  | Continuous               | Mild                   | Mild                 | Mechanical EQ     | No                          | 1     |     |            |             |</p>
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**Emergency Diesel Generator Area HVAC System**

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**Table 3.11-2 (28 of 51)**
### Fuel Handling Area HVAC System

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### Auxiliary Building Controlled Area HVAC System

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### Building Controlled Area HVAC System

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Table 3.11-2 (31 of 51)

3.11-57

Rev. 3
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**Hydrogen Monitoring System**

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**Plant Chilled Water System**

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**Essential Chilled Water System**

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### APR1400 DCD TIER 2

Table 3.11-2 (33 of 51)

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**Electric System**

<p>| PF-SW01A | 4.16kV Metal Clad Switchgear | Power Supply (PS) | AB | 078-A25A | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PF-SW01B | 4.16kV Metal Clad Switchgear | Power Supply (PS) | AB | 078-A25B | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PF-SW01C | 4.16kV Metal Clad Switchgear | Power Supply (PS) | AB | 078-A25C | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PF-SW01D | 4.16kV Metal Clad Switchgear | Power Supply (PS) | AB | 078-A25D | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PG-LC01A | 480V Load Center | Power Supply (PS) | AB | 078-A25A | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PG-LC01B | 480V Load Center | Power Supply (PS) | AB | 078-A25B | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PG-LC01C | 480V Load Center | Power Supply (PS) | AB | 078-A25C | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PG-LC01D | 480V Load Center | Power Supply (PS) | AB | 078-A25D | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |
| PG-MC01A | 480V Motor Control Center | Power Supply (PS) | AB | 100-A12A | Continuous | Mild | Mild | Electrical EQ | No | 1 | SR | O | |</p>
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**Instrumentation and Control System**

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## APR1400 DCD TIER 2

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### APR1400 DCD TIER 2

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3.11-68

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Table 3.11-2 (45 of 51)

3.11-71

Rev. 3
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(1) See EQP Table 3 for definition of environmental categories.
(2) Equipment located within a cabinet (2) is qualified allowing for temperature increase inside cabinet (2).
(3) Non-metallic consumable parts (as O-Ring, Packing and Gasket) are contained.
(4) Radiation environmental qualification requirements for individual components are developed as discussed in Subsection 3.11.5.
(5) Only Channels A and B are qualified for accident environment.
(6) EQP Table 3 provides the worst case upper bound radiation environment in the region where the component is located.
(7) RO, and TO/EQ consoles include ESF-CCS soft control modules (EPCM) which are Class 1E devices. SS and STA consoles include EPCMs which are Class 1E devices and QIAS-N FPDs which are Non-Class 1E devices. Safety Console includes Class 1E and Non-Class 1E devices. Class 1E devices are QIAS-P FPD, operator modules, Class 1E switches, EPCM. Non-Class 1E devices are QIAS-N FPDs.
Remote shutdown console includes EPCMs which are Class 1E devices and QIAS-N FPDs which are Non-Class 1E devices.
(8) QIAS-N Cabinet (2) and DRCS Remote I/O cabinets are designed by the associated circuit in accordance with the IEEE Std.384, since it performs a non-safety function, but the hardware is qualified.
(9) Piping design determines room number that valves or other equipment will be installed in and is applied to graded approach. Therefore, room number specified in the "E" column will be defined after piping design is completed. COL applicant will provide information on the room numbers (COL 3.11(7)).
(10) For cables or cable assemblies which do not have tag number due to being supplied as bulk, "N/A" will be specified in the Column labeled "Tag No".
(11) Influence of immersion means susceptible to flooding.
(12) Equipment contains electric components such as semi-conductors or organic materials. It will be designated as H(Harsh) in the Designation. If it is located in a Room with TID greater than 10 Gy.
Notes

(1) Pressure and temperature are increasing within 10 seconds from 0 psig and 49 °C (120 °F), respectively.
(2) After 182 days, pressure and temperature can be extended up to 1 year with the same values.
(3) Rapid drop within 20 seconds at 250 seconds.

Figure 3.11-1  Design Basis Containment Atmosphere Temperature and Pressure EQ Profile for Accident
Temperature (°F) 104 140 330 330 360 360 350 250 104

Notes

(1) Temperature is decreasing up to 104 °F at 86,400 seconds (24 hours)

Figure 3.11-2  Main Steam Valve Room MSLB Temperature Profile
Notes

(1) Temperature profile above applies to the following zones: 055-A46B, 068-A06A, 078-A40B, 078-A42B, and 078-A43B

Figure 3.11-3  Auxiliary Building HELB Temperature Profile
Notes

(1) Temperature profile above applies to the following zones: 078-A15C/D

Figure 3.11-4  Turbine Driven AF Pump Room Temperature Profile
Notes

(1) Temperature profile above applies to the following zones: 078-A17C/D, and 137-A30C/D

Figure 3.11-5  Turbine Driven AF Pump Room Vent & Main Steam Enclosures Temperature Profile
Figure 3.11-6  HELB HVAC Duct Tray Pipe Way Temperature Profile
Notes

(1) Temperature profile above applies to the following zones: 120-A14A and 137-A19A

Figure 3.11-7  S/G Blowdown Flash Tank Room Temperature Profile
3.12 Piping Design Review

3.12.1 Introduction

This section covers the design of the APR1400 piping systems and piping supports consisting of seismic Category I, Category II, and Category III (non-seismic). The seismic classifications are described in Subsection 3.2.1.

This section addresses the adequacy of the structural integrity as well as the functional capability of the safety-related piping system, piping components, and their associated supports. The design of piping systems provides reasonable assurance that they will perform their safety-related functions under normal operating conditions, system operating transients, postulated pipe breaks, seismic events, and combinations.

This design boundary includes pressure-retaining piping components and their supports, instrument lines, and the interaction of seismic Category II piping and associated supports with seismic Category I piping and associated supports.

This section also covers the design transients and resulting loads and load combinations with appropriate specified design and service limits for seismic Category I piping and piping supports, including those designated as ASME Section III, Class 1, 2, and 3, and those covered by ASME B31.1 and B31.3 (References 1, 2, and 3).

3.12.2 Codes and Standards

Applicable codes and standards used in the design, fabrication, construction, testing, and in-service inspection (ISI) of the piping systems and piping supports are consistent with the codes and standards specified in 10 CFR 50.55a; the applicable General Design Criteria (GDC 1, 2, 4, 14, and 15) for nuclear power plants; 10 CFR Part 50, Appendix A, as described in Section 3.1; and 10 CFR Part 50, Appendix S.

3.12.2.1 ASME Boiler and Pressure Vessel Code

The safety-related piping system design and analysis for the APR1400 are performed in accordance with the 2007 Edition with 2008 addenda of the ASME Section III (Reference 1).

However, for socket weld leg dimensions, ASME Section III, Footnote 11 to Figure NC/ND-3673.2(b)-1 in the 1989 Edition is used for socket weld with leg size less than
1.09 tₐ instead of Footnote 13 from 2007 Edition and 2008 Addenda to Figures NC/ND-3673.2(b)-1. For ASME Class 1 weld leg dimensions, the requirements of subparagraphs NB-3683.4(c)(1) and NB-3683.4(c)(2) are not applied for welds with leg size less than 1.09tₐ.

For ASME Class 1 piping, the material and D₀/t requirements of NB-3656(b) are met for all service limits when service limits include reversing dynamic loads, and alternative rules for reversing dynamic loads are used. When applying Note (1) of Figure NB-3222-1 for Level B service limits, the calculation of Pₐ stresses includes reversing dynamic loads (including inertia earthquake effects) if evaluation of these loads is required by NB-3223(b).

The non-safety-related piping system design and analysis for the APR1400 are performed in accordance with the 2010 Edition of ASME B31.1 (Reference 2) and the 2010 Edition of ASME B31.3 (Reference 3).

As described in Subsection 3.12.6, all pipe supports are designed in accordance with Subsection NF of the 2007 Edition with 2008 addenda of the ASME Section III.

### 3.12.2.2 ASME Code Cases

ASME Code Cases applicable for the piping systems and pipe supports of the APR1400 are Code Cases N-122-2, N-71-18, and N-249-14 (Reference 4).

Other ASME Code cases may be used if they are conditionally or unconditionally approved in NRC RG 1.84 (Reference 5).

### 3.12.2.3 Piping System Design Specification and Design Report

The design specification for all ASME Class 1, 2, and 3 piping systems including loading combinations, design data, and other design inputs is to be developed in accordance with ASME Section III. The design specification defines the code and the edition to be applied to the piping system design. In addition, ASME Section III requires that design reports for all ASME Class 1, 2, and 3 piping systems demonstrating and documenting that as-built piping system and pipe support configurations adhere to the requirements of the design specification (Reference 6).
3.12.3 Piping Analysis Methods

Seismic analysis methods used for all seismic Category I, seismic Category II, and non-seismic piping systems, as described below, include the response spectrum method, time-history method, or where applicable, the equivalent static load method in accordance with Subsection 3.7.3.1.1.

This subsection covers the procedure used for analytical modeling, selection of frequencies, damping criteria, combination of modal responses, the analysis for small-bore piping, and interaction of seismic Category I piping systems with other piping systems.

3.12.3.1 Experimental Stress Analysis Method

For the APR1400, experimental stress analysis methods are not used for the design of the piping system and its supports.

3.12.3.2 Modal Response Spectrum Method

3.12.3.2.1 General

The modal response spectrum method is a measure of how the piping system with certain natural frequencies responds to an earthquake applied at its pipe supports. To determine the piping system natural frequencies, each piping system is idealized as a mathematical model consisting of lumped masses connected by elastic members.

The response spectra are applied to the piping system at locations of structural attachment, such as pipe supports or equipment for each of the three orthogonal spatial components. The response spectra analysis is performed using either uniform support motion (USM) or independent support motion (ISM) method.

The response spectra analysis for piping systems uses the damping values specified in NRC RG 1.61 (Reference 7). No combination of piping damping values in NRC RG 1.61 is allowed.

3.12.3.2.2 Floor Response Spectrum

As described in Subsection 3.7.2.5, a floor response spectrum is a curve that represents the peak acceleration responses versus frequencies of a series of single-degree-of-freedom spring mass systems, which are excited by an earthquake time-history motion.
To account for the uncertainty in the structural frequencies due to uncertainties in the material properties of structures and soil and to approximations in the seismic analysis modeling techniques, the floor response spectra are smoothed with the peak values broadened in accordance with NRC RG 1.122 (Reference 8).

### 3.12.3.2.3 Uniform Support Motion Method

For analyzing the piping systems or components supported at multiple locations within a single structure or multiple structures, a uniform response spectrum (URS) that envelops all of the individual response spectra at the various support locations is used to calculate the maximum inertial responses of the piping system or component, which is defined as the uniform support motion (USM) method. The enveloped response spectrum is developed and applied in the two mutually perpendicular horizontal directions and in the vertical direction. Typically, the USM method can result in considerable overestimation of seismic responses.

The modal and spatial combination methods using the USM method are described in Subsections 3.12.3.2.4 and 3.12.3.2.5.

The piping system analyses performed using the USM method use damping values specified in Table 3 or the frequency-dependent damping values of Figure 1 of NRC RG 1.61.

### 3.12.3.2.4 Modal Combination

The response of individual modes is calculated and combined with the other modal responses using the double sum equation as described in NRC RG 1.92 (Reference 9).

\[
R_{pl} = \left[ \sum_{i=1}^{n} \sum_{j=1}^{n} \epsilon_{ij}R_{pi}R_{pj} \right]^{1/2}
\]

Where:

\[
R_{pl} = \text{combined periodic response for the } l^{th} \text{ component of seismic input motion (} l = 1, 2, \text{ and } 3, \text{ for one vertical and two horizontal components)},
\]

\[
\epsilon_{ij} = \text{the modal correlation coefficient for mode } i \text{ and } j,
\]
R_{pi}, R_{pj} = \text{periodic response or periodic components of a response of mode i and j respectively,}

n = \text{number of modes considered in the combination of modal responses.}

For the piping system modes with no closely spaced, the representative maximum responses are obtained by taking the square root of the sum of the squares (SRSS). If modes with closely spaced frequencies exist, the SRSS method is not applicable. The definition of modes with closely spaced frequencies is a function of the critical damping ratio and is as follows:

a. For critical damping ratios ≤ 2%, modes are considered closely spaced if the frequencies are within 10% of each other (i.e., for f_i < f_j, f_j ≤ 1.1 f_i)

b. For critical damping ratios > 2%, modes are considered closely spaced if the frequencies are within five times the critical damping ratio of each other (i.e., for f_i < f_j and 5% damping, f_j ≤ 1.25 f_i, for f_i < f_j and 10% damping, f_j ≤ 1.5 f_i)

If modes with closely spaced frequencies exist, the double sum equation is used considering modal correlation coefficient defined in NRC RG 1.92 (Reference 9).

The responses of low-frequency modes are obtained from all the low-frequency modes with frequencies up to the ZPA cutoff frequency.

Piping system modes greater than the ZPA cutoff frequency are considered as high-frequency or rigid range modes. The response from high frequency must be included in the response of the piping high-frequency rigid mode.

The PIPESTRESS program uses the left-out-force (LOF) method to calculate the effect of the high-frequency rigid modes. The LOF method is described in the PIPESTRESS Theory Manual (Reference 10).

PIPESTRESS generates a pseudo-load vector called a “left-out-force” vector. Generated left-out-force unit solutions are combined to approximate the contribution of the uncalculated modes to the piping system. Each left-out-force unit solution is multiplied by a scalar amplitude equal to the highest spectral acceleration for frequencies greater than the ZPA cutoff frequency. Combine the individual left-out-forces and combine the total
left-out-force response with the combination of modal responses by absolute combination, as follows:

\[ R = \left| R_{mod} \right| + \left| R_{lof} \right| \]

Where:

- \( R \) = combination of response in the periodic modes and residual rigid modes
- \( R_{mod} \) = response in the periodic modes
- \( R_{lof} \) = response in the rigid modes

The ANSYS computer program normally uses the SRSS method to combine the modal responses. In order to combine the modal response of the piping system with closely spaced modes, the double sum equation in NRC RG 1.92 (Reference 9) is used. The residual rigid responses of missing mass modes are accounted for by using the missing mass method in NRC RG 1.92 (Reference 9).

3.12.3.2.5 Directional Combination

The responses due to each of the three orthogonal spatial components of earthquake motion are combined by SRSS as described in Regulatory Position C.2.1 of NRC RG 1.92 (Reference 9).

3.12.3.2.6 Seismic Anchor Motion Analysis Method

Seismic anchor motion (SAM) analysis is a static analysis and includes the following effects acting on the piping system supported by either a single structure or more than one structure.

a. Building seismic movements

b. Equipment seismic movements as anchor motions on the piping system

c. Header piping seismic movements for decoupled branch lines

The effects of SAM on the piping system are considered for the safe shutdown earthquake (SSE).
In the SAM analysis, the relative displacements at the support are considered. The maximum relative support displacements are obtained from the structural response calculations. The support displacements are then imposed on the supported item in the most unfavorable combination by static analysis procedures.

In case of the seismic analysis using the USM method, the effects of SAMs in each of the three orthogonal spatial components are analyzed separately considering all dynamic pipe supports to be active. These interspatial responses are combined by the SRSS method to obtain cumulative effect of pipe support displacements.

The effects of these seismic anchor motions are then considered by combining with SSE inertia by the absolute summation method.

In case of the seismic analysis using the ISM method, the effects of SAMs are analyzed with the method specified in Section 2 of NUREG-1061, Volume 4 (Reference 11). The responses from SAM of each group for each spatial direction are combined by the absolute summation method. These interspatial responses are combined by the SRSS method to obtain the cumulative effect of pipe support displacements. The effects of these seismic anchor motions are considered by combining with SSE inertia using the SRSS method.

3.12.3.3 Independent Support Motion Method

The ISM method may be used in lieu of the USM method when piping systems are supported by more than one supporting structure or at multiple levels within the same structure because the USM method can result in considerable overestimation of seismic responses.

In the analysis using the ISM method, the supports are divided into support groups. Each support group consists of the supports that have similar time-history inputs. Typically, all supports in each support group are located on the same floor or portions of the floor of a structure.

The responses caused by each support group are combined by the absolute summation method. The modal and directional responses are then combined as described in Section 2 of NUREG-1061 Volume 4 (Reference 11).

Analyses performed using the ISM method are used for damping values identified in Table 3 of NRC RG 1.61 (Reference 7).
3.12.3.4 Time-History Method

The time-history method may be used for other types of dynamic analyses such as hydraulic transient loads caused by water or steam hammer, safety and relief valve discharge actuation loads, jet force loads, postulated pipe breaks, or any other dynamic loading associated with fluid flow transients.

The dynamic analysis methods used for the reactor coolant loop (RCL) piping are the direct integration method and the complex frequency response method as described in Appendix 3.9B. The surge line is evaluated using the direct integration method. For the dynamic response of the other piping systems, the time-history analysis is performed using the modal superposition method (Reference 12).

When the modal superposition method is used, the cutoff frequency for the determination of modal properties is selected to account for the principal vibration modes of the piping system based upon mass and stiffness properties, modal participation factors, and the frequency content of the input forcing function. As required on a case-by-case basis, the analysis is repeated with more modes to verify the cutoff frequency for the determination of modal properties.

The missing mass effects of high-frequency modes are included based on the same principles described in Subsection 3.12.3.2.4.

Alternatively, the cutoff frequency is determined so that the calculated number of modes produces dynamic analysis results within 10 percent of the results of the dynamic analysis, including the next higher mode.

Damping values are described in Subsection 3.12.5.4.

3.12.3.5 Inelastic Analysis Method

For the APR1400, inelastic analysis methods are not used for the design of the piping system and its supports.

3.12.3.6 Small-Bore Piping System Method

A small-bore piping system is defined as ASME Class 1 piping less than or equal to nominal diameter (DN) 25 (nominal pipe size [NPS] 1) and smaller, and the other classes of piping with nominal diameter DN 50 (NPS 2) and smaller. These small-bore piping
systems are analyzed using the response spectrum method or the equivalent static load method.

The modal response spectrum method is described in Subsections 3.12.3.2 and 3.12.3.3, and used when the equivalent static method cannot be justified.

The equivalent static load method is applied based on SRP 3.9.2, II.2.A(ii) (Reference 29). The masses of piping, its contents, and any in-line component are considered as lumped masses at their center of gravity locations. The static forces for the equivalent static load method are determined by multiplying the contributing mass by a seismic acceleration (g factor) at each location. To obtain an equivalent static load of a small-bore piping system, which is represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. This static force analysis is performed for all three spatial components of seismic excitation, and the results of these static analyses are combined by the SRSS method.

3.12.3.7 Non-Seismic/Seismic Interaction (II/I)

The APR1400 is designed to minimize the interactions of seismic Category I piping systems with non-seismic Category I piping system and to protect seismic Category I piping systems from adverse interactions with a non-seismic Category I piping system. A non-seismic Category I piping system whose continuous function is not required but whose failure could adversely affect the safety function of structures, systems, and components (SSCs) is seismically designed. The primary method of protecting a seismic Category I piping system is its isolation from a non-seismic Category I piping system. If isolation of the seismic Category I piping system is not feasible or practical, adjacent non-seismic Category I piping is classified as seismic Category II and analyzed in accordance with the same seismic design criteria applicable to the seismic Category I piping and pipe supports.

Similarly, for non-seismic piping attached to seismic piping, the dynamic effects of the non-seismic piping account for the modeling of the seismic piping. The attached non-seismic piping up to the analysis boundary is classified as seismic Category II and designed to preclude causing failure of the seismic piping during a seismic event. Non-seismic piping classified as seismic Category II is isolated from the continuing non-seismic piping by the analysis boundary, such as seismic boundary anchor. The seismic boundary anchor provides reasonable assurance that the seismic response of the seismic Category II is not affected by the dynamic effect from the non-seismic side of the anchor. The seismic
boundary anchors are designed considering the plastic hinge moment from the non-seismic piping.

3.12.3.8 Seismic Category I Buried Piping

The APR1400 design has no seismic Category I buried piping

3.12.4 Piping Modeling Technique

3.12.4.1 Computer Codes

The following computer programs are used in the analysis of seismic Category I piping designated as ASME Class 1, 2, and 3, and non-ASME piping systems. These computer programs are further described in Subsection 3.9.1.2. The applicable computer programs are as follows:

a. PIPESTRESS

PIPESTRESS is a piping analysis program that is used for the analysis of ASME Class 1, 2, and 3 as well as ASME B31.1 and B31.3 piping systems. This program is described in Subsection 3.9.1.2.1.12.

b. ANSYS

ANSYS is used in numerous applications for all components in the areas of structural, fatigue, thermal, and eigenvalue analysis including static and dynamic; elastic, plastic, creep and swelling; small and large deflections; steady-state and transient heat transfer and fluid flow. This program is described in Subsection 3.9.1.2.1.6.

c. RELAP5/MOD3.3

RELAP5/MOD 3.3 is developed by the NRC is for best-estimate transient simulation of light water reactor (LWR) coolant systems during postulated accidents in the LWR. This program is also used for the analysis of a dynamic behavior, such as water hammer and safety/relief valve discharge, by modeling the fluid flow. This program is described in Subsection 3.9.1.2.1.15.

d. GTSTRUDL
GTSTRUDL is used for structural analysis of pipe supports in conformance with ASME Section III, NF, and ANSI/AISC 360 (Reference 13). This computer program is a general-purpose structural analysis program including the base plate flexibility, anchor bolts check, and the calculation of weld leg sizes.

e. RELAP5/MOD3.1

RELAP5/MOD3.1 is a best-estimate system code suitable for the analysis of all transients and postulated accidents in LWR systems. This code is used to analyze rapid transients such as pipe breaks and valve quick opening, by modeling the fluid flow. This program is described in Subsection 3.9.1.2.1.14.

3.12.4.2 **Dynamic Piping Model**

For dynamic analysis, the seismic Category I piping systems and pipe supports are modeled as a three-dimensional space framework using a linear finite element analysis program. The analysis model consists of a series of nodes connected by beam elements with stiffness properties representing the piping and other components. Node points are modeled at points that define piping geometries as well as lumped mass locations, support locations, flange locations, expansion joint locations, or locations of structural or load discontinuities; other appurtenance locations of interest along the piping system are modeled as a node point. The RCL piping is included in RCS structural model described in Appendix 3.9B.

In the dynamic mathematical model, the distributed mass of the piping system, including pipe, contents, and insulation weight, is represented as lumped masses placed at each node.

To provide reasonable assurance that there is a sufficient number of mass points for an accurate dynamic model in the PIPESTRESS program (Reference 14), the length, \( L \) (m (ft)), is determined by the following formula based on a simply supported pipe element with fundamental natural frequency.

\[
L^4 = \left( \frac{\alpha^2}{f_R^2} \right) (EI / W)
\]

Where:

\[
E = \text{modulus of elasticity} \ [\text{N/mm}^2 (\text{psi})]
\]

\[
I = \text{moment of inertia} \ [\text{mm}^4 (\text{in}^4)]
\]
If the distance between mass points exceeds L/2 when the automatic mass modeling is used, then additional mass points are generated.

Concentrated weights of the components, such as valves, flanges, and other appurtenances, are also modeled as lumped masses. The torsional effect due to eccentricity of an eccentric mass, such as a valve operator, is considered.

In general, pipe supports are modeled as rigid with the rigidity verified by checking support deflection in the restrained direction, if springs with actual stiffness values for the restrained degrees of freedom. Pipe support hardware weight for snubbers, struts, and spring hangers supported by the piping system is considered in the piping analysis. The weight added by the component support is included in the piping analysis when it is greater than 10 percent of the total mass of the adjacent pipe span including pipes, contents, insulations, and in-line components.

In general, an entire piping system cannot be modeled and analyzed as a single model; the piping system is therefore conveniently divided into multiple, smaller piping subsystems that satisfy the analysis size limitations of the computer program used for the piping system analysis. Branch piping that does not have a significant effect on the run piping is decoupled from the run pipe analysis based on the branch decoupling criteria defined in Subsection 3.12.4.4. Intermediate pipe anchors such as wall or slab penetration sleeve anchors and structural anchor supports may also be used for subdividing the piping systems.

3.12.4.3 Piping Benchmark Program

The computer programs used for the piping system analysis are verified in accordance with NRC benchmark problems.

The piping benchmark problems prescribed in NUREG/CR-1677, Volumes 1 and 2 (Reference 15), are used to validate the PIPESTRESS and ANSYS computer programs used in piping system analysis.
3.12.4.4 Decoupling Criteria

Small branch lines including instrument connections may be decoupled from the analysis model of the larger run pipe provided that the ratio of the moments of inertia of the two lines ($I_b/I_r$) is less than or equal to 1/25, under such conditions that no restraints on the branch are located near the run pipe for the flexibility or no precise magnitudes of reactions are required at the terminal points (Reference 28).

In the case that the branch piping geometry is known, piping systems are consider in accordance with the decoupling criteria described in Subsection 3.7.2.3.2.

In the run pipe analysis, the applicable stress intensification factors (SIFs) and/or stress indices are incorporated. The mass effects of the branch line, where the mass of half the span of the branch pipe is greater than 10 percent of the mass of the pipe run span, are also considered at the run pipe connection point. If a large valve or other large concentrated mass is located within the first span of the branch piping, the torsional effects of the eccentric mass is considered. In these cases, the branch pipe is included in the run pipe model up to the first anchor or four seismic supports in each of the three perpendicular directions to consider eccentric mass effect.

In the decoupled branch pipe analysis, the run pipe connection point is modeled as an anchor with the same SIF and/or stress indices as the run pipe and the effect of seismic excitation from run pipe is considered as follows. The movements of the run pipe due to the thermal, seismic anchor motion (SAM), or pipe break analyses at the header connection are applied as anchor motions in the branch line analysis. In addition, seismic inertial movement from the run pipe analysis is also considered at the header connection. The inertial effects of the run pipe on the branch line are considered in the branch piping analysis to capture the possible amplification of inertial input from the run pipe. The effects of seismic excitations from the two seismic supports in each of the three perpendicular directions to the branch connections in the run pipe are included in the branch piping analysis.

3.12.5 Piping Stress Analysis Criteria

3.12.5.1 Seismic Input Envelope vs. Site-Specific Spectra

Seismic input envelope and site-specific spectra of the APR1400 are described in Subsection 3.7.2.5.
3.12.5.2 Design Transients

RCS design transients used for the design and fatigue analysis of ASME Class 1 piping systems and support components are addressed in Table 3.9-1.

3.12.5.3 Loadings and Load Combination

3.12.5.3.1 Pressure

Internal design pressure, \( P \), is used in the design and analysis of ASME Class 1, 2, and 3 piping (Reference 1). Minimum pipe wall thicknesses are determined using the formulations of NB/NC/ND-3640 and the design pressure. The applicable design and maximum service level pressures are used in load combinations as identified in Tables 3.12-1 and 3.12-2.

3.12.5.3.2 Mechanical Loads

The weight of the piping system, its contents, any insulation and in-line equipment, and any other mechanical loads identified in the design specification are considered in the piping analysis. The weight of water during hydrostatic testing is considered for steam or air-filled piping systems.

3.12.5.3.3 Thermal Expansion

The effect of linear thermal expansion range during various operating modes is considered along with thermal movements of terminal equipment nozzles, anchors, or restraints (thermal anchor movements) corresponding to the operating modes. The stress-free temperature is taken as 21\(^{\circ}\)C (70\(^{\circ}\)F). The piping systems operating at a temperature of 65\(^{\circ}\)C (150\(^{\circ}\)F) (Reference 1) and below are not analyzed for the effects of linear thermal expansion.

3.12.5.3.4 Seismic

The effects of seismic inertial loads and anchor movements are included in the design analysis. The ground motion of the operating basis earthquake (OBE) for the APR1400 is equal to one-third of the ground motion of the SSE. Per Appendix S to 10 CFR Part 50, the OBE load case does not require explicit design analysis. In the event of an earthquake that meets or exceeds the OBE ground motion, plant shutdown is required and seismic Category I piping and supports are inspected to provide reasonable assurance that no
functional damage has occurred. The design of the APR1400 seismic Category I piping and supports includes analysis of the inertial and anchor movement effects of the SSE event. These loads are Service Level D loads.

Fatigue effects due to earthquake loads are addressed in Table 3.12-1. Tables 3.12-1 and 3.12-2 identify SSE inertial and displacement loads in various load combinations for ASME Class 1, 2, and 3 piping and piping supports.

3.12.5.3.5 Fluid Transient Loads

The relief/safety valve thrust loads for open or closed systems are functions of valve opening, flow rate, flow area, and fluid properties. The analysis of these loads is usually accomplished using static loads as input to the piping analysis with appropriate dynamic load factors. Dynamic analysis of relief valve thrusts is used when static analysis produces undesirably conservative results. These loads are considered in Service Level B or D load combinations.

The water hammer phenomenon involves the rapid change in fluid flow creating a “shock wave” effect in the piping system. They are usually set in motion by rapid actuation of control valves, relief valves, and check valves. Rapid start or trip of a pump or turbine can also initiate such a phenomenon. The water hammer phenomenon is analyzed using dynamic analysis methods. The water hammer loads are considered in Level B, or D service load combinations.

The fluid transient loads are identified as dynamic fluid loads in Tables 3.12-1 and 3.12-2.

3.12.5.3.6 Wind/Tornado Loads

Safety-related piping systems of ASME Class 1, 2, and 3 are designed within the wind/tornado protected structure. If COL applicant finds it necessary to route ASME Class 1, 2, or 3 piping systems outside the structure, the wind and/or tornado load must be included in the plant design basis loads considering the site-specific loads (COL 3.12(1)).

3.12.5.3.7 Pipe Break Loads

High-energy line breaks cause loads in the form of pipe whip, jet impingement, and changes in environmental conditions. Pipe break loads include the impact of the RCPB piping break, main steam and feedwater line breaks except for piping breaks that meet the
leak-before-break (LBB) criteria (see Subsection 3.6.3) or inside the pipe break exclusion area. Pipe break loads are considered in Level D service load combinations.

3.12.5.3.8 Thermal and Pressure Transient Loads

Thermal and pressure transients are evaluated in the analysis of ASME Class 1 piping by calculating the range of primary plus secondary stress intensities. For ASME Class 2 and 3 piping, these transients are included as load cases in the appropriate ASME Code equations.

3.12.5.3.9 Hydrostatic Pressure Tests

Piping systems are tested for leaks by filling the system with the test fluid and pressurizing to test pressures specified in the design specification. Piping systems that normally carry operating fluids, such as steam or gas, have stops placed in spring hangers and temporary supports added as needed. The effects of the test pressure and fluid weight are considered in satisfying the appropriate ASME Class 1, 2, and 3 stress equations. The effects of hydrostatic pressure tests on ASME Class 1 piping fatigue are in accordance with NB-3226.

3.12.5.3.10 Load Combinations

Using the methodology and equations from the ASME Code, pipe stresses are calculated for various load combinations. The ASME Code includes design limits for design conditions; Service Levels A, B, C, and D; and testing. Load combinations for ASME Class 1 piping are given in Table 3.12-1. ASME Class 2 and 3 load combinations are given in Table 3.12-2.

3.12.5.4 Damping Values

Damping values in Table 3 of NRC RG 1.61 (Reference 7) are used for dynamic response spectra and time-history analyses.

Frequency-dependent damping values identified in Figure 1 of NRC RG 1.61 may also be used for USM response spectra analysis provided the five restrictions identified in C.2 of NRC RG 1.61 (Reference 7) are maintained.
3.12.5.5 Combination of Modal Responses

Seismic responses to each mode are calculated in accordance with the method described in NRC RG 1.92 (Reference 9) and combined with other responses. Seismic responses to periodic modal response with sufficiently separated frequencies are combined by SRSS. Closely spaced frequencies are combined by the double sum equation described in Subsection 3.12.3.2.4.

3.12.5.6 High-Frequency Modes

PIPESTRESS computer program uses left-out-force (LOF) method to calculate the effect of high-frequency rigid modes (Reference 10). The result obtained from this method is multiplied by scalar amplitude that is equivalent to the highest spectral acceleration for frequencies, which is greater than the last natural frequency being calculated by LOF method regarding the corresponding directional spectrum. ANSYS computer program accounts for the effects of high frequency modes in accordance with the missing mass method in NRC RG 1.92 (Reference 9).

3.12.5.7 Fatigue Evaluation of ASME Code Class 1 Piping

Fatigue evaluation of ASME Class 1 piping systems is performed for loadings caused by thermal and pressure transients, thermal stratification, and other cyclic events including earthquakes. Fatigue evaluation of ASME Class 1 piping greater than DN 25 (NPS 1) is performed per ASME Section III, NB-3653.

The fatigue evaluation considering the effects of the reactor coolant environment in ASME Class 1 piping follows the guidance in NRC RG 1.207 (Reference 17).

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

The calculation for the cumulative usage factors of ASME Class 2 and 3 piping is not required. Fatigue evaluation of ASME Class 2 and 3 piping is not performed in accordance with the requirements in NC/ND-3653.2(a). Acceptable cyclic stress is reduced by applying stress range reduction factor, f, to thermal expansion stress ranges in accordance with Table NC/ND-3611.2(e)-1. The stress intensification factors that are applicable to piping components and joints are based on fatigue testing.
3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

Unisolable sections of piping connected to the reactor coolant system (RCS) that could be subjected to stresses from thermal stratification caused by valve leakages or turbulent penetrations are reviewed to provide reasonable assurance of the structural integrity of the lines.

APR1400 conforms with the requirements in U.S. NRC Bulletin 88-08 (Reference 18) for all piping connected to the reactor coolant system. Data available from the reference plant have been evaluated and incorporated into the design of the APR1400.

Based on the temperature distributions in the piping between the direct vessel injection (DVI) nozzle and the first isolation valve, and the piping between shutdown cooling system (SCS) nozzle and the first isolation valve, which were evaluated using a commercial thermal-hydraulic analysis code, it is expected that the temperature difference in stratified flow is relatively small and the thermal stratification effects would be negligible for the SCS suction line.

The effect of thermal stratification on the piping system is analyzed in two parts:

a. Global stratification, which causes bending

b. Local stratification, which causes stresses similar to thermal gradient stresses

To consider the global stratification, the global bending stress due to a vertical temperature distribution in horizontal and non-horizontal members is mathematically defined in the computer program PIPESTRESS (Reference 14) as follows:

\[
g_u = \frac{\int T(v)vd\Omega}{I_u}
\]

Where:

\[\Omega = \text{the pipe wall region}\]

\[T(v) = \text{a vertical temperature distribution on pipe cross-section}\]
\[ I_a = \text{the moment of inertia} \]
\[ v = \text{vertical direction against axis of a horizontal pipe} \]

Then, \( \alpha g_a \) is the load on the pipe due to the temperature distribution \( T(v) \), where \( \alpha \) is the coefficient of thermal expansion of the pipe. The unit of \( \alpha g_a \) is rotation-per-unit-length.

Local stresses due to thermal stratification are confined to thermal stratification zone. The boundary conditions are such that local thermal stresses are due only to temperature distribution across the pipe wall. The local stress intensity due to thermal stratification is calculated for the fatigue analysis of the ASME Class 1 piping system. The steps for piping fatigue analysis are as follows:

a. Thermal-hydraulic analysis

b. Conversion of thermal-hydraulic analysis data

c. Direct application of temperature distribution data into three-dimensional piping model

d. Extraction of the maximum stress intensity (local stress)

e. Evaluation of code requirements in accordance with ASME Section III, NB-3600

The reactor vessel and piping segments away from the stratified zone are not affected by thermal stratification and are not considered in local stress determination. The maximum stress intensity extracted from three-dimensional finite element analysis is added to the \( \Delta T_2 \) term of ASME Section III, NB-3653.2 Equation (11) without considering stress index.

3.12.5.10 Thermal Stratification

NRC Bulletin 79-13 (Reference 19) addresses the effect of thermal stratification that leads to the cracking of the feedwater line. The APR1400 feedwater lines are designed to minimize the thermal stratification. The feedwater lines are angled downward from the horizontal to minimize the potential for thermal stratification. This is also described in Subsection 5.4.2.1.2.1.3.

NRC Bulletin 88-11 (Reference 20) was issued in response to the results of an inspection of the surge line at Trojan that showed large, unexpected movements that closed available
The APR1400 pressurizer surge line features and operational procedures address the structural integrity issues raised by NRC Bulletin 88-11 in minimizing surge line stratification.

a. The surge line piping is routed to insure that sufficient flexibility exists to minimize resistance to uniform thermal growth and the thermal deformation due to stratified flow loadings. The surge line has the horizontal section between the vertical take-off at top of the hot leg and vertical leg at the bottom of the pressurizer. The horizontal section is sloped downward at a minimum rate of 1/16 inch per foot from the pressurizer.

b. The nozzle and surge line from the hot leg are upward vertical and of sufficient length to prevent the colder hot leg fluid from entering the surge line beyond the take-off. During the normal power operation, a continuous bypass spray flow is maintained to further suppress turbulent penetration from the hot leg flow.

c. The maximum temperature difference between the pressurizer and the hot leg can occur during plant heatup and cooldown. The maximum temperature difference is determined based on the shutdown cooling system (SCS) alignment condition. During plant heatup, the RCS is pressurized by the bubble formation using the pressurizer heaters until the system pressure is high enough for RCP operation. The pressurizer temperature increases along with the saturation temperature as the system pressure increases. For the determination of the maximum temperature difference, the pressurizer is conservatively assumed to be heated up to the maximum system pressure for SCS alignment. This temperature difference is established as the design criteria for the surge line and the system operating procedure for heatup and cooldown.

d. The pressurizer versus hot leg temperature differential is limited during heatup and cooldown below the design temperature differential of 188.9 °C (340 °F). The differential temperature can be determined by the difference between the pressurizer and the hot leg temperature indications. The differential temperature
limits are specified in the operating procedures. Also, the pressurizer heaters are used to maintain a required pressurizer and temperature during the initial heatup. The pressurizer level is maintained at a predetermined level (outsurge) during the plant heatup by the charging and letdown flow controls, which can limit the cooler hot leg fluid insurge.

The APR1400 design conforms with NRC Bulletin 88-11 pressurizer surge line requirements as follows:

a. The stratification test for the surge line piping layout for Younggwang Nuclear Power Plant Unit 3 (YGN 3) was done as part of the YGN 3 pre-core hot functional test. The measurements taken during the test verified the assumptions and methodology of the surge line design basis both from a thermal-hydraulic and structural standpoint. The measurements included both the circumferential and axial temperature gradients and the pipe displacements. The surge line for APR1400 has the almost same operational procedure and piping layout as those for YGN 3 of OPR1000 and was also tested during Shin-Kori Nuclear Power Plant Unit 3 (SKN 3) pre-core hot functional test. Only the pipe displacements were measured from the SKN 3 test because it was confirmed from the YGN 3 test that there are the relationships between pipe temperatures, curvatures, and displacements which forms the basis of the design analysis. The pipe displacements measured from the SKN 3 test verified the design of surge line for APR1400.

b. The COL applicant will implement the monitoring program during the first preoperational testing and continue to monitor by using the fatigue monitoring system during the first cycle of operation to verify the design transients used in the structural design of the surge line (COL 3.12(2)). The monitoring program includes the real-time measurements of the surge line pipe displacements and temperatures at several major locations on the surge line and plant data such as hot leg and pressurizer temperatures, pressurizer pressure and level, charging and letdown flows as well as the status of reactor coolant pumps, pressurizer heaters, and spray valves.

Structural evaluations are performed using elastic and/or simplified elastic-plastic analyses in accordance with the ASME Code, considering the applicable loadings in addition to the stratified flow loadings.
The stratified-flow-induced curvature of the piping and local stresses due to a temperature gradient are obtained from finite element analyses. These analyses provide the local effects and pipe rotations for an unsupported pipe segment. A stratified flow thermal-hydraulic model with the top half of the fluid at hotter temperature and the lower half of the fluid at colder temperature is used to determine the pipe wall temperature based on the thermal-hydraulic conditions. Heat transfer and structural thermal stress analyses are performed using the ANSYS computer program to determine the rotations and local stresses. Rotations are considered to act over all horizontal portions of the pipe. The resulting bending moments are calculated in the piping analysis with the ANSYS computer program by allowing the pipe to thermally expand unconstrained and by then applying a set of equal and opposite displacements at the rigid support points. Local stress effects due to top-to-bottom thermal gradients are also considered to act over all horizontal sections of pipe. For ASME Class 1 piping, gross bending stresses due to stratification are considered as secondary stresses, while local stresses due to thermal gradients are considered as peak stresses.

3.12.5.11 Safety Relief Valve Design, Installation, and Testing

The design and installation of the safety valves and relief valves for overpressure protection are performed per the requirements in Appendix O of the ASME Code (Reference 21).

A static method with a conservative dynamic loading factor is used to calculate the discharge forces of safety valves and relief valves that use open vent stacks for discharging fluid directly into the air. Dynamic transient loads of fluid discharged from safety/relief valve to vessels or headers are considered as forces acting on the changes in direction such as elbows or branch connections. Piping stresses and support/restraint loads resulting from these discharge forces of safety and relief valves are assessed by dynamic time-history analysis or by an equivalent static force analysis in piping systems.

See Subsection 3.12.4.1.c for the computer program used in the analysis.

3.12.5.12 Functional Capability

Functional capability of all ASME Class 1, 2, and 3 piping systems essential for safe shutdown of the plant is reasonably assured in accordance with NUREG-1367 (Reference 22) to provide sufficient fluid flow path under Service Level D loading conditions.
3.12.5.13 **Combination of Inertial and Seismic Anchor Motion Effects**

The inertial and seismic anchor motion (SAM) effects are analyzed separately. The results of SAM analysis are combined with those of seismic inertial analysis by absolute summation when an enveloped USM method is used, per SRP 3.7.3, and by the SRSS when the ISM method is used for the dynamic analysis per NUREG-1061, Volume 4 (Reference 11).

3.12.5.14 **Operating Basis Earthquake as a Design Load**

The applicable earthquake load in the design of APR1400 piping systems is described in Section 3.7. The operating basis earthquake (OBE) ground motion has been set as one-third of SSE and, therefore, is not considered in the seismic design in accordance with 10 CFR Part 50.

3.12.5.15 **Welded Attachments**

If integral welded attachments are used to attach supports on piping systems, they are evaluated by the nonmandatory Appendix Y of ASME Section III, “Evaluation of the Design of Rectangular and Hollow Circular Cross Section Welded Attachments on Class 1, 2, and 3 Piping.”

3.12.5.16 **Modal Damping for Composite Structures**

The composite modal damping for coupled building and piping systems is used for piping systems that are coupled to concrete building structures, if applicable. The procedure used to determine the composite modal damping value for the piping system is described in Subsection 3.7.2.15.

3.12.5.17 **Minimum Temperature for Thermal Analyses**

The stress-free state temperature for a piping system is typically defined as 21 °C (70 °F). The analysis for a piping system with an operational temperature of greater than 65 °C (150 °F) or less than 4 °C (40 °F) is performed for the effect of thermal expansion/contraction.

3.12.5.18 **Intersystem Loss-of-Coolant Accident**

The design feature of low-pressure piping systems that interface with reactor coolant pressure boundary (RCPB) is described in Appendix 5A.
3.12.5.19 Effects of Environment on Fatigue Design

The fatigue evaluation considering the effects of the reactor coolant environment in ASME Class 1 piping follows the guidance in NRC RG 1.207 (Reference 17).

3.12.6 Piping Support Design Criteria

This section provides piping support design methods, procedures and criteria, and piping support design criteria provided in Subsection 3.9.3 are used as references.

3.12.6.1 Applicable Codes

Seismic Category I pipe supports are designed in accordance with ASME Section III, NF for Service Levels A, B, C, and D, and the acceptance limits of Appendix F of ASME Section III for Service Level D.

Standard component supports are designed, manufactured, installed, and tested pursuant to NF of ASME Section III.

For non-seismic category pipe supports supporting piping analyzed to ASME B31.1, the requirements of ASME B31.1 for supports (Sections 120 and 121) are met, where applicable. In addition, the structural elements are designed using guidance from the AISC 360 (Reference 13).

In addition to the pipe support design codes mentioned above, expansion anchors and other steel embedments in concrete are designed in accordance with Subsection 3.12.6.4.

3.12.6.2 Jurisdictional Boundaries

The jurisdictional boundary between the pipe and its support structure follows the guidance of NB-1132, NC-1132, or ND-1132, as appropriate for the ASME Section III Class of piping involved.

The jurisdictional boundary between the pipe support and the building structure follows the guidance of ASME Section III, NF-1130. In general, for attachments to building steel, the boundary is taken at the interface with the building steel, with the weld being designed to the rules of NF. For attachments to concrete building structures, the boundary is generally at the weld of the support member to a baseplate or embedded plate, with the weld again being designed to the rules of NF.
3.12.6.3  Loads and Load Combinations

Loading conditions and load combinations for piping supports are defined in Table 3.9-10. Load combinations for piping support design used for Service Levels A, B, C, and D include piping reaction loads calculated for load combinations given in Table 3.12-1 and Table 3.12-2. The deadweight of the support itself for Service Levels A, B, C, and D, friction loads (Subsection 3.12.6.10) for Service Levels A, B, C, and D seismic self-weight excitations (Subsection 3.12.6.8) for Service Level D are considered in addition to piping reaction loads.

The stress limits for pipe support designs meet the criteria of ASME Section III, NF.

3.12.6.4  Pipe Support Baseplate and Anchor Bolt Design

Although the use of baseplates with expansion anchors is expected to be minimized in the APR1400 design, baseplate designs are likely to be needed. For these designs, the concrete is evaluated using ACI 349-01, Appendix B (Reference 23), subject to the conditions and limitations of NRC RG 1.199 (Reference 24). This guidance accounts for the proper consideration of anchor bolt spacing and distance to a free edge of concrete. In addition, all aspects of the anchor bolt design, including baseplate flexibility and factors of safety, will be used in the development of anchor bolt loads, as addressed in NRC Bulletin Letter 79-02 (Reference 25).

3.12.6.5  Use of Energy Absorbers and Limit Stops

Energy absorbers and limit stops for pipe supports are not used for the APR1400 design.

3.12.6.6  Use of Snubbers

Snubbers for piping systems are used for situations requiring free thermal movements, while restraining movements due to dynamic loadings. Typical snubber components are manufactured standard hardware, and may be either hydraulic or mechanical in operation.

Snubbers are not capable of supporting gravity loads. Under certain circumstances, the weight of the snubber bearing on the pipe is included in the piping stress analysis. Snubbers in general are not to be used where thermal movements are small. Also, use of snubbers is minimized to the extent that is reasonable due to the maintenance and testing requirements for these components. As such, accessibility of any snubbers that are used is considered in the design of the piping system.
The functional requirements and design specifications provided to the supplier of snubbers contain the information described in Subsection 3.9.3.4.

3.12.6.7 **Pipe Support Stiffness**

Rigid stiffness is used for the piping supports in the piping analysis model with a check on support deflection in the restrained directions to verify the rigidity. The actual stiffness is modeled for variable spring supports. If the actual support stiffness is used for any support other than variable spring supports, all supports within the piping model use the actual support stiffness.

Each support modeled as rigid is checked with the deflection in the restrained directions to a maximum of 1.6 mm (1/16 in.) for SSE loadings, and a maximum of 3.2 mm (1/8 in.) for other loadings.

3.12.6.8 **Seismic Self-Weight Excitation**

The excitation of the support structure to SSE loadings is to be included in the pipe support analysis. Damping values for welded and bolted structures are given in NRC RG 1.61 (Reference 7). This support self-weight SSE response and the piping inertial load SSE response are to be combined by absolute summation.

3.12.6.9 **Design of Supplementary Steel**

This subsection provides design information on any supplementary steel required to connect the main support structure to the building structure.

As addressed in Subsection 3.12.6.1, all seismic Category I pipe supports for the APR1400 are designed to ASME Section III, NF. For non-seismic pipe supports, AISC 360 (Reference 13) is used for the supplementary steel, as it is for the main support structure.

3.12.6.10 **Consideration of Friction Forces**

Friction forces are developed in the pipe support when a pipe slides across the surface of a support member in the unrestrained directions under thermal expansion conditions. Because friction is due to the movement of the pipe, loads from friction are calculated using the deadweight and applicable signed loads normal to the applicable support member.
Specifically, the friction forces need to be calculated only if the thermal movement in the applicable unrestrained directions is greater than 1.6 mm (1/16 in.). The coefficient of friction is taken as 0.3 for steel-to-steel conditions and 0.1 for low-friction slide/bearing plates.

3.12.6.11 Pipe Support Gaps and Clearances

For guide type pipe supports modeled as rigid restraints in the piping analysis, the typical industry design practice is to provide small gaps between the pipe and its surrounding structural members. These small gaps allow radial thermal expansion of the pipe as well as allow rotation of the pipe at the support. The normal design practice for the APR1400 is to use a nominal cold condition gap of 1.6 mm (1/16 in.) on each side of the pipe in the restrained direction. For vertical restraint supports in horizontal piping, the pipe will be in contact with the support member in the direction of gravity. The gap between the upper side of the pipe and the support is allowed to have a range up to 3.2 mm (1/8 in.). If frame type supports are used for the larger diameter (>24 in) high temperature (>350 °F) piping, the gap is calculated considering temperature and pressure to allow free radial expansion of the piping and the gap is specified on the pipe support design drawing.

3.12.6.12 Instrumentation Line Support Criteria

The design and analysis loadings, load combinations, and acceptance criteria to be used for instrumentation line supports are similar to those used for pipe supports. The applicable design loads include deadweight, thermal expansion, and seismic loadings where appropriate. The applicable loading combinations similarly follow those used for the ASME Section III Levels in Table 3.9-10 using the design loadings mentioned above. The acceptance criteria are in accordance with ASME Section III, NF for seismic Category I instrumentation lines, AISC 360 (Reference 13) for non-seismic instrumentation lines.

3.12.6.13 Pipe Deflection Limits

For standard component pipe supports using standard manufactured hardware components, the manufacturer’s recommendations for limitations in its hardware are followed. The limitations are travel limits for spring hangers; stroke limits for snubbers; swing angles for rods, struts, and snubbers; alignment angles between clamps or end brackets with their associated struts and snubbers; and the variability check for variable spring supports. In addition to the manufacturer’s recommended limits, allowances are made in the initial
designs for tolerances on such limits. This is especially important for snubber and spring design in which the function of the support may be changed by an exceeded limit.

3.12.6.14 Clamp-induced Local Pipe Stress Evaluation

Stiff pipe clamps of the type identified in Information Notice (IN) 83-80 and Generic Safety Issue (GSI) 89 are not used for the piping supports.

3.12.7 Graded Approach of Piping Systems

The ASME Code Class 1 piping systems include RCS main loops, pressurizer surge line and thirteen piping subsystems of RCS branch piping which include four DVI lines, two SC lines, and four RCS drain lines (letdown, charging, RG and spray lines). Based on the safety function, integrity, piping size and layout, the RCS main loops, pressurizer surge line, DVI and SC lines are selected as ASME Code Class 1 for the graded approach. The DVI and SC lines are selected as the representative RCS branch piping systems based on size; other lines are smaller diameter and have no significant impact on RCS integrity. Considering the symmetrical arrangement of the DVI and SC lines, one out of four SI lines and one out of two SC lines were chosen as representative cases.

The scope of design for ASME Code Class 2 and 3 piping includes main steam and main feedwater piping located inside the containment building. Main steam and main feedwater piping is the largest ASME Class 2 and 3 piping connected to the steam generators and has the highest structural load. The scope of design for main steam and main feedwater piping outside the containment building covers piping (one division) from the containment penetration anchors to the main steam valve house (MSVH) penetration anchors beyond the isolation valves which are located in the break exclusion area in auxiliary building.

The results of the piping analyses are documented in Technical Reports, APR1400-E-B-NR-16001-P, APR1400-E-B-NR-16002-P and APR1400-H-N-NR-14005-P (References 30, 31 and 32), and are commensurate with the requirements of regulations (10 CFR 50.55A; 10 CFR Part 50, Appendix A, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15; and 10 CFR Part 50, Appendix S).
Combined License Information

COL 3.12(1) If COL applicant finds it necessary to route ASME Class 1, 2, or 3 piping systems outside the structure, the wind and/or tornado load must be included in the plant design basis loads considering the site-specific loads.

COL 3.12(2) The COL applicant will implement the monitoring program during the first preoperational testing and continue to monitor by using the fatigue monitoring system during the first cycle of operation to verify the design transients used in the structural design of the surge line.

References


<table>
<thead>
<tr>
<th>Service Condition</th>
<th>Service Level</th>
<th>Category</th>
<th>Loading</th>
<th>Acceptance Criteria&lt;sup&gt;(1)&lt;/sup&gt;</th>
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<tbody>
<tr>
<td>Design</td>
<td>-</td>
<td>Primary Stress</td>
<td>Design pressure, deadweight, steady-state flow load&lt;sup&gt;(2)&lt;/sup&gt; specified as level A</td>
<td>Eq. 9 NB-3652 1.5 $S_m$</td>
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<tr>
<td>Normal /Upset</td>
<td>A/B</td>
<td>Primary plus Secondary Stress Intensity Range (S.I.R.)</td>
<td>Service pressure, steady-state flow load, dynamic fluid load&lt;sup&gt;(2)&lt;/sup&gt;, thermal expansion load&lt;sup&gt;(3)&lt;/sup&gt;, thermal expansion anchor motion load&lt;sup&gt;(3)&lt;/sup&gt;, cyclic thermal load&lt;sup&gt;(4)&lt;/sup&gt;, material discontinuity stress, earthquake inertial load&lt;sup&gt;(7)&lt;/sup&gt;, IRWST discharge load</td>
<td>Eq. 10 NB-3653.1 3 $S_m$</td>
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<td>Peak S.I.R.</td>
<td>Service pressure, steady-state flow load, dynamic fluid load&lt;sup&gt;(2)&lt;/sup&gt;, thermal expansion load&lt;sup&gt;(3)&lt;/sup&gt;, thermal expansion anchor motion load&lt;sup&gt;(3)&lt;/sup&gt;, cyclic thermal load&lt;sup&gt;(4)&lt;/sup&gt;, material discontinuity stress, earthquake inertial load&lt;sup&gt;(7)&lt;/sup&gt;, IRWST discharge load, thermal radial gradient stress (linear and nonlinear)</td>
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<td>Thermal S.I.R.&lt;sup&gt;(5)&lt;/sup&gt;</td>
<td>Thermal expansion load&lt;sup&gt;(3)&lt;/sup&gt;, thermal expansion anchor motion load&lt;sup&gt;(3)&lt;/sup&gt;, cyclic thermal load&lt;sup&gt;(4)&lt;/sup&gt;</td>
<td>Eq. 12 NB-3653.6(a) 3 $S_m$</td>
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<td>Primary plus Secondary Membrane plus Bending S.I.R.&lt;sup&gt;(5)&lt;/sup&gt;</td>
<td>Service pressure, steady-state flow load, dynamic fluid load&lt;sup&gt;(2)&lt;/sup&gt;, material discontinuity stress, earthquake inertial load&lt;sup&gt;(7)&lt;/sup&gt;, IRWST discharge load</td>
<td>Eq. 13 NB-3653.6(b) 3 $S_m$</td>
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<td>Alternating Stress Intensity (S.I.) (Fatigue Usage)&lt;sup&gt;(6)&lt;/sup&gt;</td>
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<td>NB-3653.3 NB-3653.4 NB-3653.5 NB-3653.6(c)</td>
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<td>Permissible Pressure</td>
<td>Maximum level D service pressure</td>
<td>NB-3656(a)</td>
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<td>Primary Stress</td>
<td>Coincident level D service pressure, deadweight, steady-state flow load, dynamic fluid load, earthquake inertial load, high-energy line break load, IRWST discharge load</td>
<td>NB-3656(a)</td>
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<td>Secondary Stress</td>
<td>Max [range of (bending moments due to thermal expansion load plus thermal expansion anchor motion load plus ½ earthquake anchor motion load) or range of earthquake anchor motion load]</td>
<td>NB-3656(b)</td>
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<td>Primary Membrane S.I.</td>
<td>Test pressure, deadweight</td>
<td>NB-3657</td>
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<td>Primary Membrane plus Bending S.I.</td>
<td>Test pressure, deadweight</td>
<td>NB-3657</td>
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</table>
Table 3.12-1 (3 of 4)

(1) Acceptance criteria are taken from the referenced section in Section III of the ASME Code or are as noted.

(2) Dynamic fluid loads are occasional loads associated with hydraulic transients caused by events such as valve actuation (safety or relief valve discharge, rapid valve opening/closing), water hammer, or steam hammer.

(3) Thermal expansion and thermal expansion anchor motion loads are not calculated for those operating conditions where the piping system does not exceed 65 °C (150 °F).

(4) Cyclic thermal load includes loads due to thermal stratification, and stresses due to high-cycle thermal striping and thermal penetration (i.e., thermal mixing).

(5) The thermal bending and primary plus secondary membrane plus bending stress intensity ranges (Equations 12 and 13) are only calculated for those load sets that do not meet the primary plus secondary stress intensity range (Equation 10) allowable.

(6) The cumulative fatigue usage factor is calculated by summing the Level A and Level B fatigue usage. If applicable, pressure testing conditions is also included in the calculation of the cumulative usage factor (see Note13).

(7) The earthquake inertial load considered in the Level B primary plus secondary stress intensity range, peak stress intensity range and alternating stress intensity calculations (Equations 10, 11, and 14) is taken as one-third of the peak SSE inertial load or as the peak SSE inertial load. If the earthquake inertial load is taken as the peak SSE inertial load, then 20 cycles of earthquake loading are considered. If the earthquake inertial load is taken as one-third of the peak SSE inertial load, then the number of cycles to be considered for earthquake loading is as described in Subsection 3.7.3.1.2 (the equivalent number of 20 full SSE cycles as derived in accordance with Appendix D of IEEE Std. 344 (Reference 26 in Subsection 3.12.9)).

(8) The resultant moment calculated is the maximum of the resultant moment due to the full range of earthquake inertial load or the resultant moment due to the consideration of half of the range of earthquake inertial load with all other applicable loads.

(9) The rules given in Appendix F of the ASME Code may be used in lieu of those given in NB-3656(a) and NB-3656(b) when evaluating Level D primary stress.

(10) Loads due to dynamic events other than high-energy line break (i.e., loss-of-coolant accident and secondary side pipe rupture) and SSE are combined considering the time phasing of the events (i.e., whether the loads are coincident in time). When the time phasing relationship can be established, dynamic loads may be combined by the square-root-sum-of-the-squares (SRSS) method, provided it is demonstrated that the non-exceedance criteria given in NUREG-0484 (Reference 16 in Subsection 3.12.9) are met. When the time phasing relationship cannot be established, or when the non-exceedance criteria in NUREG-0484 are not met, dynamic loads are combined by absolute sum. SSE and high-energy line break loads are always combined using the SRSS method.
(11) This secondary stress check is only necessary if the stresses (including those due to earthquake inertial load) exceed the Equation 10 (primary plus secondary stress intensity range for the upset service condition) allowable stress. See Section NB-3656(b)(4) in Section III of the ASME Code.

(12) \( S_m \) = Allowable design stress intensity value from Part D of Section II of the ASME Code.

(13) If a piping system is subjected to more than 10 pressure test cycles that result in an alternating stress intensity \( (S_a) \) value greater than that for 106 cycles, as determined from the applicable fatigue design curves of Figures I-9.0 in Section III of the ASME Code, then those cycles in excess of 10 are included in the fatigue calculation that determines the cumulative usage factor. See NB-3657 and NB-3226(e) in Section III of the ASME Code.
### Table 3.12-2

Loading Combinations for Acceptance Criteria for ASME Section III, Class 2 and 3 Piping

<table>
<thead>
<tr>
<th>Service Condition</th>
<th>Service Level</th>
<th>Loading</th>
<th>Acceptance Criteria(^{(4)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design</td>
<td>-</td>
<td>Pressure, Weight, Other Sustained Mechanical Loads</td>
<td>Eq. 8 NC/ND-3652(^{(3)})</td>
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<tr>
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<td>1.5 (S_h) (^{(3)})</td>
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<tr>
<td>Normal / Upset</td>
<td>A/B</td>
<td>Pressure, Weight, Other Sustained Mechanical Loads, Dynamic Fluid Loads (DFL)(^{(1)}), IRWST discharge load</td>
<td>Eq.9 NC/ND-3653.1 (Level B Only) (^{(6)})</td>
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<td></td>
<td>Thermal Expansion, Thermal Anchor Movement (TAM)</td>
<td>Eq.10 NC/ND-3653.2(a)(^{(2)})</td>
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<td>Building Settlement</td>
<td>Eq. 10a NC/ND-3653.2(b)</td>
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<td></td>
<td>Pressure, Weight, Other Sustained Mechanical Loads, Thermal Expansion, TAM</td>
<td>Eq. 11 NC/ND-3653.2(c)(^{(2)})</td>
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<td>Emergency</td>
<td>C</td>
<td>No loads. Refer to Subsection 3.9.3.1.</td>
<td>Eq. 9 NC/ND-3654.2(a)(^{(5)})</td>
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<tr>
<td>Faulted</td>
<td>D</td>
<td>Pressure, Weight, DFL(^{(1)}), SSE Inertia, Pipe Break (^{(8)}), IRWST discharge load</td>
<td>Eq. 9 NC/ND-3655(a)(^{(5)})</td>
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<td></td>
<td>Thermal Expansion, TAM, Seismic Anchor Motion (SAM)</td>
<td>NC/ND-3655(b)(^{(4)})</td>
</tr>
</tbody>
</table>

(1) Dynamic fluid loads (DFLs) are occasional loads such as safety/relief valve thrust, steam hammer, water hammer, or other loads associated with plant upset, emergency, or faulted conditions as applicable.

(2) Stresses are to meet the requirements of either Equation 10 or 11, not both.

(3) If, during operation, the system normally carries a medium other than water (air, gas, steam), sustained loads are to be checked for weight loads during hydrostatic testing as well as normal operation weight loads.

(4) ASME Section III.

(5) When causal relationships can be established, dynamic loads may be combined by SRSS, provided it is demonstrated that the non-exceedance criteria given in NUREG-0484 are met. When the causal relationship cannot be established, or when the non-exceedance criteria given in NUREG-0484 are not met, dynamic loads are to be combined by absolute sum. SSE and high-energy line break loads are always combined using the SRSS method.

(6) OBE inertia and SAM loads are not included in the design of Class 2 and 3 piping (Reference 27 in Subsection 3.12.9).


(8) Pipe break loads include loads due to LOCA.
3.13 Threaded Fasteners (ASME Section III, Class 1, 2, and 3)

This section addresses relevant requirements of GDC 1, 4, 14, 30, and 31, and 10 CFR Part 50 Appendices B and G for ASME Section III (Reference 1), Class 1, 2, and 3 component threaded fasteners.

ASME Section III, Class 1, 2, and 3 component fasteners materials are selected, fabricated, designed, tested, and inspected to meet the requirements of ASME Section III, NB, NC, and ND corresponding to their ASME Code Classes except reactor vessel stud bolts, for which the detailed description is provided in paragraph 5.3.1.7.

3.13.1 Design Considerations

3.13.1.1 Materials Selection

ASME Section III, Class 1, 2, and 3 component fasteners are fabricated using the materials that are prescribed in ASME Section III. Threaded fasteners, except the stud bolts for the reactor vessel head and reactor coolant pump (RCP) casing, which consist of the reactor coolant pressure boundary (RCPB) or are in contact with the primary coolant, are made of primary water corrosion-resistant materials such as austenitic stainless steels, martensitic stainless steels, precipitation-hardened stainless steels, and nickel-based alloys. In designing threaded fastener joints, consideration is given for the prevention of galvanic corrosion, except when the design or material of the fasteners has been demonstrated to be acceptable through satisfactory operation in the OPR 1000 plants, where any primary coolant leakage can be automatically identified if it occurs, or where periodic inspections for leakage and verification of the integrity of threaded fasteners are performed as a countermeasure for leakage.

Table 3.13-1 lists the applicable criteria in ASME Section III paragraphs relevant to the material selection and testing of threaded fasteners in Class 1, 2, and 3 components. Materials used in threaded fasteners are selected for their compatibility with the environmental conditions to which they are exposed.

3.13.1.2 Special Materials Fabrication Processes and Special Controls

Special process and controls such as heat treatment, which affects the material properties, and tests for material property evaluation of threaded fasteners procured or fabricated, are in accordance with ASME Section II (Reference 2) and III. Table 3.13-1 identifies the
appropriate NB-2200, NC-2200, or ND-2200 of the ASME Code regarding material heat treatment and tensile test coupons preparation criteria for ferritic materials (e.g., carbon steel, high-strength low-alloy steel). The criteria of ASME Section III are applied rather than the criteria of the material specification of ASME Section II applicable to the mechanical testing if there is a conflict between the two sets of criteria.

In addition, threaded fasteners are designed and fabricated using materials and processes that have been selected to minimize stress corrosion cracking or other forms of material degradation.

Threaded fasteners are cleaned in accordance with NRC RG 1.28 (Reference 3) to provide reasonable assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking.

Reactor vessel closure studs and nuts are surface-treated with a manganese-based phosphate coating for protection against corrosion effects. The reactor vessel closure studs and nuts are fabricated to meet 10 CFR Part 50, Appendix G requirements as stated in Subsection 5.3.1.7.

3.13.1.2.1 Fabrication Inspection

ASME Section III, Class 1, 2, and 3 component fasteners are inspected during fabrication in accordance with ASME Section III, NB-2580, NC-2580, and ND-2580 for the corresponding ASME Code Class (see Table 3.13-1) and material specification.

3.13.1.2.2 Lubricants and Sealants

Lubricants used during installation of ASME Section III, Class 1, 2, and 3 component fasteners are those that meet the requirements prescribed in the relevant design documents to provide reasonable assurance of the compatibility with the fasteners. Lubricants are selected in accordance with the guidance provided in NUREG-1339. Acceptable lubricants are Loctite N-5000, Neolube, and Never Seez Pure Nickel Special Nuclear Grade. These lubricants have shown satisfactory performance in OPR 1000 plants. MoS₂ is not allowed for use with code class fasteners in any circumstance. A single type of lubricant is used at the site, if possible, for easy control. Sealants are not used for the threaded fasteners.
Reactor vessel closure studs and nuts are surface treated with manganese-base phosphate coating for protection against the possibility of incurring corrosion effects.

3.13.1.3 Fracture Toughness Requirements for Threaded Fasteners Made of Ferritic Materials

ASME Section III, Class 1, 2, and 3 component fasteners with diameters that are greater than 25 mm (1 in.) are impact tested in accordance with ASME Section III, NB-2300, NC-2300, or ND-2300 (as appropriate for the assigned ASME Code Class) at a temperature no higher than the lowest service temperature or preload temperature, whichever is less, and meet the acceptance criteria described in ASME Section III, NB-2330, NC-2330, or ND-2330 as for the corresponding Code Class (see Table 3.13-1).

In addition, ferritic bolts, studs, and nuts (i.e., bolts, studs, and nuts made from either low-alloy steel or carbon steel materials) used in RCPB applications meet the fracture toughness requirements of 10 CFR Part 50, Appendix G.

3.13.1.3.1 Prevention of Brittle Fracture and Failure due to Fatigue and Structural Stress

ASME Section III, Class 1, 2, and 3 component fasteners are designed in accordance with ASME Section III, NB-3000, NC-3000, and ND-3000, respectively, to endure the stresses resulting from operation of the corresponding components where the fasteners are installed. In addition, they are evaluated for fatigue and are designed to have enough toughness to prevent brittle fracture during the operation in accordance with ASME Section III, NB-3000, NC-3000, or ND-3000 as for the corresponding code class.

3.13.1.4 [Reserved]

3.13.1.5 Certified Material Test Reports

Quality records such as certified material test reports (CMTRs), which are associated with ASME Section III, Class 1, 2, and 3 component fasteners, are controlled, maintained, and stored in accordance with the quality assurance program to meet the requirements of 10 CFR Part 50, Appendix B; ASME NQA-1; and ASME NCA. The CMTRs contain the contents required by ASME Section III, NCA-3860. Documentation related to fracture toughness testing results for ASME Code Class 1, 2, and 3 component threaded fastener...
materials, when required, is to be maintained as part of QA records in compliance with the requirements of the ASME Section III.

The combined license (COL) applicant is to maintain quality assurance records including CMTRs on ASME Section III, Class 1, 2, and 3 component threaded fasteners in accordance with the requirements of 10 CFR 50.71 (COL 3.13(1)).

3.13.2 Inservice Inspection Requirements

As required by 10 CFR 50.55a, except where written relief has been granted by the NRC, the relevant requirements of ASME Section XI, Division 1 (Reference 4) are followed for the plant preservice and inservice inspection of ASME Section III, Class 1, 2, and 3 threaded fasteners (see Table 3.13-2).

The COL applicant is to submit the preservice and in-service inspection program for ASME Section III, Class 1, 2, and 3 component threaded fasteners to the NRC prior to performing the inspections (COL 3.13(2)). The inservice inspection program identifies the applicable edition and addenda of ASME Section XI and provides reasonable assurance of conformance with the requirements of 10 CFR 50.55a (b) (2) (xxvi).

3.13.3 Combined License Information

COL 3.13(1) The COL applicant is to maintain quality assurance records including CMTRs on ASME Section III, Class 1, 2, and 3 component threaded fasteners in accordance with the requirements of 10 CFR 50.71.

COL 3.13(2) The COL applicant is to submit the preservice and inservice inspection programs for ASME Section III, Class 1, 2, and 3 component threaded fasteners to the NRC prior to performing the inspections.

3.13.4 References


### Table 3.13-1

**ASME Section III Criteria for Selection and Testing of Bolting Materials** (1)

<table>
<thead>
<tr>
<th>Code Category</th>
<th>ASME Class 1 Components</th>
<th>ASME Class 2 Components</th>
<th>ASME Class 3 Components</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material Selection</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heat treatment criteria</td>
<td>NCA-1220 and NB-2128</td>
<td>NCA-1220 and NC-2128</td>
<td>NCA-1220 and ND-2128</td>
</tr>
<tr>
<td>Test coupon requirements for bolting/stud materials</td>
<td>NB-2221 NB-2224</td>
<td>NC-2221 NC-2224.3</td>
<td>ND-2221 ND-2224.3</td>
</tr>
<tr>
<td>Fracture toughness requirements</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material to be impact tested</td>
<td>NB-2311</td>
<td>NC-2311</td>
<td>ND-2311</td>
</tr>
<tr>
<td>Types of impact test</td>
<td>NB-2321</td>
<td>NC-2321</td>
<td>ND-2321</td>
</tr>
<tr>
<td>Test coupons</td>
<td>NB-2322</td>
<td>NC-2322</td>
<td>ND-2322</td>
</tr>
<tr>
<td>Acceptance standards</td>
<td>NB-2333</td>
<td>NC-2332.3</td>
<td>ND-2333</td>
</tr>
<tr>
<td>Number of impact tests necessary</td>
<td>NB-2345</td>
<td>NC-2345</td>
<td>ND-2345</td>
</tr>
<tr>
<td>Retesting</td>
<td>NB-2350</td>
<td>NC-2352</td>
<td>ND-2352</td>
</tr>
<tr>
<td>Calibration of test equipment</td>
<td>NB-2360</td>
<td>NC-2360</td>
<td>ND-2360</td>
</tr>
<tr>
<td>Examination criteria for bolts, studs, and nuts</td>
<td>NB-2580</td>
<td>NC-2580</td>
<td>ND-2580</td>
</tr>
<tr>
<td>Certified material test report criteria</td>
<td>NCA-3860</td>
<td>NCA-3860</td>
<td>NCA-3860</td>
</tr>
</tbody>
</table>

(1) Section III paragraphs listed in this table represent those specified in the 2007 Edition of ASME Section III.
### Table 3.13-2

**ASME Section XI Examination Categories for Inservice Inspections of Mechanical Joints in ASME Code Class 1, 2, and 3 Systems Secured by Threaded Fasteners**

<table>
<thead>
<tr>
<th>Examination Type</th>
<th>ASME Class 1 Components</th>
<th>ASME Class 2 Components</th>
<th>ASME Class 3 Components</th>
</tr>
</thead>
<tbody>
<tr>
<td>Specific bolting inspections</td>
<td>Table IWB-2500-1, Exam. Cat. B-G-1 for bolting greater than 5.08 cm (2 in.) in diameter</td>
<td>Table IWC-2500-1, Exam. Cat. C-D for bolting greater than 5.08 cm (2 in.) in diameter</td>
<td>Not applicable – Currently, there are no examination categories that correspond to those that exist for ASME Class 1 and 2 bolting.</td>
</tr>
<tr>
<td></td>
<td>Table IWB-2500-1, Exam. Cat. B-G-2 for bolting less than or equal to 5.08 cm (2 in.) in diameter</td>
<td></td>
<td></td>
</tr>
<tr>
<td>System pressure tests</td>
<td>Table IWB-2500-1, Exam. Cat. B-P</td>
<td>Table IWC-2500-1, Exam. Cat. C-H</td>
<td>Table IWD-2500-1, Exam. Cat. D-B</td>
</tr>
</tbody>
</table>

(1) Section XI paragraphs listed in this table represent those specified in the 2007 Edition of ASME Section XI.