3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

This section addresses methods of analysis for Seismic Category I components and supports, including those designated as ASME Code Class 1, 2, 3 (or core support) and those not covered by the ASME Code. Information is also presented concerning design transients. The following GDC apply to this section:

- GDC 1 requires that structures, systems, and components (SSC) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR 50, Appendix B requires provisions to assure that appropriate standards are specified and included in design documents—including design methods and computer programs for the design and analysis of Seismic Category I, ASME Code Class 1, 2, 3, and core support structures and non-Code structures—and that deviations from such standards are controlled. Special topics for mechanical components encompass items related to design transients (e.g., component supports, core supports, and reactor internals) that are designated as ASME Code Class 1, 2, and 3 and also those not covered by the Code.

- GDC 2 requires that SSC important to safety are 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. 10 CFR 50 Appendix S specifies that applicants include seismic events in the design basis. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena (see Section 3.2.2). Special topics for mechanical components encompass items related to design transients that are designated as Level A (Normal), Level B (Upset), Level C (Emergency), Level D (Faulted) and Test.

- GDC 14 requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and of gross rupture. GDC 15 requires that the reactor coolant system (RCS) and its auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. Safety-related mechanical components are designed to remain functional under postulated combinations of normal operating conditions, anticipated operational occurrences, postulated pipe breaks, and seismic events. Compliance with the requirements of GDC 14 and GDC 15 demonstrates that the design transients and consequent loads and load combinations (with the appropriate specific design and service limits for ASME Code Class 1 and core support components, supports, and reactor internals) form a complete basis for the design of the RCPB for anticipated conditions and for extremely low-probability events postulated to occur during the service life of the plant.

To further demonstrate compliance with the requirements of GDC 1, 2, 14, and 15, this section provides a list of transients that are used in the design and fatigue analysis of the Code Class 1 and core support components, supports, and reactor internals within the RCPB. The number of events for each transient is also included. Additionally, to demonstrate compliance with the requirements of 10 CFR 50, Appendix B and GDC 1, this section also contains a list of computer programs that are used in dynamic and static analyses.

Other sections that interface with this section are:

- Section 15.0 describes the acceptability of the transients and the number of events expected over the service lifetime of the plant.
- Section 3.12 addresses the effects of the reactor coolant environment on fatigue. Thermal stratification is also addressed in Section 3.12.
- Section 3.13 describes bolting and threaded fastener adequacy and integrity.
- Section 3.7.3 describes the seismic cyclic ground input loading and the method for determining the seismic cyclic loading used for fatigue analysis of appropriate components.
- Section 6.1.1 describes the consideration given to minimize degradation of materials due to corrosion based upon the environmental conditions to which equipment will be exposed.

3.9.1.1 Design Transients

The design transients define thermal-hydraulic conditions (i.e., pressure, temperature, and flow) for the RCPB. Bounding thermal-hydraulic design transients are defined for components of the RCPB and the secondary side pressure boundary (SSPB) with respect to mechanical behavior. The number of design transients is based on a plant life of 60 years. The transients are defined for equipment design purposes and are not intended to represent actual operating experience.

The following operating conditions, as defined in the ASME Boiler and Pressure Vessel Code, Section III (Reference 1) apply to the RCS, RCS component supports, and reactor pressure vessel (RPV) internals:

- 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 anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which 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 those deviations from normal conditions which 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. If the total number of postulated occurrences over the plant design lifetime for such events exceeds 25 strong stress cycles, they are evaluated for cyclic fatigue using Level B service limits. Strong stress cycles have an alternating stress value greater than that associated with 10^6 cycles from the applicable fatigue design curves in Section III of the ASME Code.

- Faulted conditions (ASME Service Level D).

  Faulted conditions are those 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 compliance with safety criteria. The methods of analysis to calculate the stresses and deformations conform to the methods outlined in the ASME Code, 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. The first 10 hydrostatic tests, the first 10 pneumatic tests, or any combination of 10 such tests do not need to be considered in the fatigue evaluation of the components or piping in accordance with ASME Code requirements.

Table 3.9.1-1—Summary of Design Transients lists the design transients and the number of events for each transient. The load combinations and their acceptance criteria are provided in Section 3.9.3 and Appendix 3F. The transients listed in Table 3.9.1-1 are assumed for the design life of the plant. In accordance with the ASME Code, Section III, emergency and faulted conditions are not included in fatigue evaluations, with the exception that any significant emergency cycles in excess of 25 must be considered in the fatigue analyses. Significant emergency cycles are those that result in stresses higher than the endurance limits on the ASME design fatigue curves.

The transient conditions selected for equipment fatigue evaluation are based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. The transients selected are representative of operating conditions which are considered to occur during plant operation and are severe or frequent enough to be of possible significance to component cyclic behavior and fatigue life. The transients selected are a conservative representation of transients which, when used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant. The term “stretch-out operation” refers to the maximum fuel cycle length based on an assumed T_{AVG} reduction of 10°F, followed by a coastdown to 70 percent full power. This corresponds to approximately 40 effective full-power days (EFPD) at the end of the fuel cycle.

Although the U.S. EPR will be operated as a base-loaded plant, the reference U.S. EPR design provides robust features for the effects of load follow. Similarly, the structural design and analysis of the RCS, RCS components, RCS component internals, and systems ancillary to the RCS account for the effects of load follow.

3.9.1.1.1 Normal Conditions

The following RCS transients are considered normal conditions:

3.9.1.1.1.1 Plant Heatup and Cooldown

Heatup transients are analyzed for initial conditions following the reactor pressure vessel (RPV) head being removed (typically due to a refueling outage) or for a cold shutdown (CSD) condition where the RPV head has not been removed. Similarly, cooldown transients are also analyzed with these conditions as final conditions.

The following Normal transients represent U.S. EPR heatup and cooldown operations:

Transient 1 - Plant Startup from CSD to Full Load with the RPV Open at CSD

Transient 1A - Complete Plant Startup from CSD to Full Load, with RPV Open at CSD

This transient is based on normal operations during a complete plant startup from cold shutdown following a refueling outage to full load. During the startup that follows a short outage, the RPV is heated from ambient to 122°F by residual heat before starting the RCPs. The RCS is heated to hot shutdown (HSD) at an upper rate limit of 72°F per hour followed by an increase to 100 percent full power (FP).

The temperature gradient is maximized during the heatup phase since the maximum gradient should be lower than 72°F/hr based on the expected residual heat from the core and RCP heat.

Transient 1B - Plant Startup from CSD to Full Load with the RPV Closed at CSD

This transient is based on normal operations during a complete plant startup from cold shutdown to full load. This transient is identical to normal transient 1A except that the initial RCS temperature is 122°F.

Transient 2 - Complete Plant Shutdown from Full Load to CSD

Transient 2A - Plant Shutdown from Full Load to CSD with RPV Open

This transient is based on normal operations during a complete plant shutdown from full load to HSD and then to cold shutdown with the RPV open for refueling. The rate of power decrease is 5 percent FP per minute. The RCS cooldown continues to a final RCS temperature of 60°F.

Transient 2B - Plant Shutdown from Full Load to CSD with RPV Closed

This transient represents the normal operations during a complete plant shutdown from full load to HSD and then to CSD without removal of the reactor head. This transient is identical to Normal transient 2A except that the reactor head is closed at CSD and the final RCS temperature is 122°F.

Transient 3 - Heatup to 250°F with Subsequent Shutdown

This transient consists of a heatup of the RCS from CSD to 250°F followed by a return to CSD. The heatup and cooldown are identical to those described in normal transients 1A and 2A except that the maximum RCS temperature is 250°F.

Transient 4 - Shutdown to 250°F with Subsequent Startup

This transient consists of a power ramp from 100 percent FP to HSD followed by a cooldown of the RCS to 250°F with the steam generators (SG) removing heat. After a stabilization phase at the intermediate shutdown state, the RCS is heated followed by a power increase to full load. This transient encompasses a reactor trip with a loss of the process instrumentation and control system (PICS) and a reactor trip with PICS available.

3.9.1.1.1.2 Unit Loading and Unloading

The following Normal transients represent unit loading and unloading between various power levels.

Transient 5 – Power Ramps from 100 percent FP to 0 percent FP at 5 percent FP/min and Back

Transient 5A - Power Ramp from Full Load to HSD at 5 Percent of FP per Minute

This transient addresses load ramps between power operation and HSD during both automatic and manual phases at low load. It does not include load follow transients (Normal transient 6 includes load follow). The ramp rate is 5 percent FP per minute. The first phase of this transient is a power decrease from 100 percent FP to a reduced power level < 25 percent FP. The second phase continues the power decrease at a rate of 5 percent FP per minute until HSD is reached.

Transient S5A - Power Ramp from Full Load of Stretch-Out Operation to HSD at 5 Percent FP per Minute

This transient is identical to Normal transient 5A except that it addresses load ramps between power operation in stretch-out operation and HSD during both automatic and manual phases at low load.

Transient 5B - Power Ramp from HSD to Full Load at 5 Percent FP per Minute

This transient includes load ramps between HSD and FP operation during both automatic and manual phases at low load. It does not include load follow transients (Normal transient 6 includes load follow). A rate of 5 percent FP per minute is used for this transient.

Transient S5B - Power Ramp from HSD to Full Load of Stretch-Out Operation at 5 Percent FP per Minute

This transient is identical to Normal transient 5B except that it addresses load ramps between HSD and power operation in stretch-out operation using automatic and manual phases at low load. A rate of 5 percent FP per minute is used for this transient.

Transient 6 - Daily Load Follow

Transient 6A - Power Ramp from Full Load to 60 Percent FP and Back to Full Load

This transient, induced by the daily load follow (fluctuation in power the plant goes through to match electrical load demand), consists of a power ramp between full load and an intermediate power level of 60 percent FP and back to FP with a rate of 5 percent FP per minute. This transient also includes a return to full load with a rate of power change of 5 percent FP per minute.

Transient 6B - Power Ramp from Full Load to 25 Percent FP and Back to Full Load

This transient is similar to Normal transient 6A except that it consists of a power ramp between full load to 25 percent FP and back with a rate of power change of 5 percent FP per minute between 60 percent FP and 100 percent FP, and 2.5 percent FP per minute between 25 percent FP and 60 percent FP.

3.9.1.1.1.3 Load Regulation and Steady-State Fluctuations

Load regulation refers to fluctuations in load due to the plant participating in some form of grid frequency control. The U.S. EPR design considers the following frequency control transients and unscheduled power variations are represented in several transients.

Reactor coolant pressure can vary around steady-state values. In addition to the load fluctuations at power, the fluctuations at HSD are represented.

Transient 7 - Remote Control/Frequency Control

Transient 7A - Primary Frequency Control

Primary frequency control corresponds to local control of turbo-generator rotation speed. The rotation speed of the turbo-generator is adjusted to match power production with power consumption, thus keeping the grid frequency at its reference value of 60Hz. The normal range for power variations caused by this control is ±2.5 percent FP load step.

Transient S7A - Primary Frequency Control in Stretch-Out Operation

This transient is the corresponding event to normal transient 7A, but in stretch-out operation.

Transient 7B - Secondary Frequency Control in ±12.5 Percent FP Load Change

Secondary or remote frequency control is used when the primary frequency control cannot maintain the grid frequency at 60Hz. This transient consists of power ramps with amplitude of ±12.5 percent FP with a rate of power change of 1 percent FP per minute. The normal range for power variations of this amplitude is between 60 percent FP and 100 percent FP where the average primary temperature is maintained constant.

Transient 8 - Unscheduled Power Variations

Transient 8A – Power Change of +10 Percent FP Followed by a Power Ramp of 5 Percent FP per Minute and Return to Starting Power

In the event of grid perturbations, the plant must be able to perform fast power variations in order to stabilize the grid. This transient defines a power change of +10 percent FP (1 percent FP per second) immediately followed by a power ramp at a rate of 5 percent FP per minute. The amplitude of the total power variation is 20 percent FP. The power level during this transient does not exceed 95 percent FP.

Transient 8B - Unscheduled Power Variations – Power Ramp at -20 Percent FP per Minute

This transient consists of a power ramp from 100 percent FP down to 25 percent FP at a rate of -20 percent FP per minute.

Transient 8C - Unscheduled Power Variations – Power Change of ±10 Percent FP at a Rate of ±1 Percent FP per Second

This transient is a power increase of +10 percent FP to 100 percent FP, or a power decrease of 10 percent FP to 30 percent FP at a rate of 1 percent FP per second.

Transient 8D - Unscheduled Power Variations – After Power Decrease, Power Ramp from 25 Percent FP to Full Load with a Power Rate of 5 percent FP per Minute

This transient represents restarts following unscheduled power variations with power decreases. These restarts are performed with a load ramp from 25 percent FP to full load with a power rate of 5 percent FP per minute. This transient reflects a return to FP following Normal transient 8B.

Transient 9 – Unscheduled/Spurious Fluctuations at HSD

This transient consists of temperature and pressure fluctuation under HSD conditions. It includes unscheduled fluctuations of small amplitude at HSD, hot standby, and low load (<10 percent FP) conditions. The amplitudes of temperature and pressure fluctuations are 10°F and 150 psi, respectively.

3.9.1.1.1.4 Partial Trip

These transients represent the U.S. EPR partial trip function which allows a step decrease in turbine load from core FP (100 percent FP) followed by a stabilization at 25 percent FP. Each of the following Normal transients includes a partial trip along with various operations that follow.

Transient 10 - Partial Reactor Power Reduction to 25 percent of Full Load

Transient 10A - Partial Trip to 25 Percent FP and Restart to Full Load

This transient corresponds to a step decrease in turbine load from FP to 25 percent FP followed by a stabilization of power at 25 percent FP, and subsequent return to FP at 5 percent FP per minute. This transient can be initiated by the loss of one RCP.

Transient S10A - Partial Trip to 25 Percent FP and Restart to Full Load in Stretch-Out Operation

This transient is similar to Normal transient 10A except that it occurs during stretch-out operation.

Transient 10B - Successful Transfer to House Load – Partial Trip to 25 Percent FP Followed by Restart to Full Load

This power reduction from full load to 25 percent FP is due to a transfer to houseload, which is the result of a fault on the grid leading to the opening of the main breaker and the resulting disconnection of the turbo-generator from the grid. When the power reduction is finished, the plant is stabilized at 25 percent FP followed by a subsequent return to FP. The power increase rates are the same as those described in Normal transient 10A.

Transient 10C - Partial Trip Followed by a Decrease to HSD and a Restart to Full Load

This transient is similar to Normal transient 10A, but does not include an immediate restart to FP after a stabilization period at 25 percent FP. Instead, after a partial trip to 25 percent FP, the power is decreased to 0 percent FP at a rate of 5 percent FP per minute. After a stabilization period at HSD conditions, the power is returned to full load at a rate of 5 percent FP per minute.

Transient 10D - Partial Trip Followed by a Decrease to CSD

This transient is similar to Normal transient 10A, but instead after a partial trip to 25 percent FP, the plant is taken to CSD conditions. The power decrease from 25 percent FP to CSD occurs at a rate of 5 percent FP per minute.

3.9.1.1.2 Upset Conditions

The following RCS transients are considered Upset conditions:

Upset Transient 1 - Reactor Trip

Upset Transient 1A - Reactor Trip at FP with Subsequent Return to FP

This transient addresses manual trips, spurious trips, or trips resulting from disturbances such as failures in feedwater control or reactivity anomalies that do not lead to temperature and pressure fluctuations before trip. In this transient, the plant is restarted after a stabilization period of three hours at HSD. Following stabilization at HSD, the plant is returned to FP.

Upset Transient S1A - Reactor Trip in Stretch-Out Operation with Subsequent Return to FP in Stretch-Out Operation

This transient is similar to Upset transient 1A except that it occurs during stretch-out operation.

Upset Transient 1B - Reactor Trip at FP Followed by a Shutdown to CSD

In Upset transients 1A and S1A, the reactor restarts to full load after stabilization at HSD conditions. In this transient the reactor trip is followed by a normal cooldown to the CSD state.

Upset Transient 2 - Turbine Trip and Subsequent Return to FP

This transient represents a turbine trip which results in SG and RCS heatup and overpressurization. The initial heatup and overpressurization of the SGs and the RCS, due to the turbine trip, is limited by the overpressure protection and by the reactor trip. After a stabilization period at HSD conditions, the reactor is restarted to FP.

Upset Transient 3 - Short-Term Loss of Power with Subsequent Return to FP

This transient represents short-term loss of power operation (i.e., loss of offsite power (LOOP), but external electrical sources are recovered before plant shutdown). This transient includes a loss of the RCPs, SGs steam, and main feedwater flow because of the LOOP, followed by a failure of the transfer to houseload. The startup of the RCPs after the restoration of offsite power causes a drop in RCS and pressurizer (PZR) temperatures. After stabilization at HSD, the plant is returned to FP.

Upset Transient 4 - Loss of Feedwater (LOFW) with Subsequent Return to FP

Upset Transient 4A - LOFW with Subsequent Return to FP (SSS Injection)

This transient is initiated by a LOFW not originating from a LOOP. This transient results in a reactor trip due to low SG level. The Startup and Shutdown Feedwater System (SSS) starts up on an LOFW pump signal or a low SG level signal. After stabilization at HSD and after recovering FW capability, the plant is restarted to FP.

Upset Transient 4B - LOFW with Subsequent Return to FP (EFWS Actuation)

Transient 4B is the same scenario as Transient 4A with actuation of EFWS on low-low SG level, when the SSS pump is assumed to be unavailable due to a common failure with main feedwater pumps.

Upset Transient 5 - Spurious RCS Depressurization with Subsequent Return to FP

This transient involves a spurious depressurization of the RCS due to a failure of the control of PZR spray valves. This leads to a depressurization of the RCS until the low pressurizer pressure reactor trip setpoint is reached. When the safety injection signal is reached, the partial cooldown begins on the secondary side. After stabilization at HSD, the plant is restarted to FP.

Upset Transient 6 - Reactor Trip with Excessive Secondary Side Heat Removal and Subsequent Return to FP

This transient results from a spurious safety injection signal that leads to a reactor trip and a partial cooldown. At the end of the partial cooldown, failure of a turbine bypass system (TBS) valve to close is assumed. SG pressure decreases until main steam isolation valve (MSIV) closure occurs. After MSIV closure, the SG pressure increases to the main steam relief train (MSRT) setpoint. After a stabilization period at HSD, the plant is restarted to FP.

Upset Transient 7 - Excessive Feedwater Supply at HSD

This transient addresses the excessive cooling of the RCS and SG at HSD conditions. This transient also addresses spurious startup of the Emergency Feedwater System (EFWS) during shutdown conditions. In this transient, full opening of one main feedwater (MFW) valve is assumed which results in overcooling of a SG and the associated RCS loop. The heat removal caused by this excess flow results in a drop in SG temperature until MFW isolation occurs. To prevent EFWS actuation, this transient assumes manual re-actuation of the MFWS and SSS. Consequently, primary and secondary side pressures and temperatures are stabilized at HSD conditions.

Upset Transient 8 - Secondary Side Depressurization Leading to a Maximum Pressure Difference between the RCPB and the SSPB

This transient addresses events leading to maximum pressure differences between the primary and secondary sides. For example, this transient event could result from a loss of power leading to a simultaneous loss of the RCPs and SG feedwater flow rates, followed by a failure of a TBS valve to close.

Upset Transient 9 - Unscheduled Fluctuations between HSD and CSD

Two categories of fluctuations are considered for this transient: low amplitude fluctuations and high amplitude fluctuations. Low amplitude fluctuation results from a control system malfunction or a manual operation. High amplitude fluctuation results from an upset transient.

Upset Transients 9A/B - Low amplitude fluctuations resulting from a control system malfunction or manual operation

The low amplitude fluctuations are caused by the following:

- RCP startup or shutdown under normal conditions.
- Fluctuations due to the RCS temperature control.
- Fluctuations due to PZR level or pressure control.
- Manual control of the SG levels.

Upset Transients 9C/D – High amplitude fluctuations resulting from an upset transient initiated between HSD and CSD conditions

The high amplitude is caused by the following:

- Maneuverability tests of PZR safety valves.
- Spurious opening of one TBS valve.
- Spurious safety injection signal.
- Startup of auxiliary systems during plant startup and shutdown (i.e., CVCS, low head safety injection (LHSI), residual heat removal (RHR), normal spray, and auxiliary spray).

The following cases of high amplitude fluctuations between HSD and CSD are considered:

- Heating in hot conditions (SG connected): Heating with the SGs not removing heat during a short period of time before operator action.
- Cooling in hot conditions (SG connected): Spurious opening of an MSRT. Two cases were considered corresponding to high initial SG pressure and low initial SG pressure.
- Pressurization in hot conditions (SG connected): Spurious startup of medium head safety injection (MHSI) pumps.

- Depressurization in hot conditions (SG connected): Spurious opening of a normal spray valve and a larger-than-expected opening of a PZR safety valve during startup tests.
- Heating in cold conditions (RHRS connected): Heating with RHRS unavailable during a short period of time before operator action. Two cases are considered corresponding to hot initial conditions and cold initial conditions.
- Cooling in cold conditions (RHRS connected): Spurious opening of an RHRS temperature control valve.
- Pressurization in cold conditions (RHRS connected): Spurious startup of MHSI pumps. Two cases are considered corresponding to hot initial conditions and cold initial conditions.
- Depressurization in cold conditions (RHRS connected): Spurious opening of a spray valve.
- Heating in solid state (RHRS connected): Heating with RHRS unavailable for a short period of time before operator action.
- Cooling in solid state (RHRS connected): Spurious opening of an RHRS temperature control valve.
- Pressurization in solid state (RHRS connected): Spurious closure of letdown line or spurious startup of MHSI pumps.
- Depressurization in solid state (RHRS connected): Spurious interruption of RCS charging.

Upset Transient 10 - Maximum SG Pressure with Open RCS

This transient consists of a test of the leak tightness of the SG with the RCS open at CSD. This test is performed in CSD conditions with primary side temperature and pressure of 140°F and 14.7 psia, respectively. The test is performed by pressurizing the SG, while keeping the RCS temperature and pressure constant under CSD conditions.

Upset Transient 11 - Inadvertent Closure of One MSIV

This transient involves the inadvertent closure of an MSIV at FP. Closure of the MSIV leads to an increase in both secondary side pressure and temperature, thus causing primary side pressure and temperature to increase. Reactor trip and turbine trip are automatically actuated and secondary pressure in the affected SG is automatically limited by MSRT operation. After a stabilization period of three hours at HSD conditions, the plant is restarted to FP.

3.9.1.1.3 Emergency Conditions

The following transients are considered Emergency conditions:

Emergency Transient 1 - LOOP with Natural Circulation Cooldown

This transient consists of a LOOP of an extended duration leading to initiation of a shutdown of the plant.

The transient consists of the loss of the RCPs and SG steam and feedwater flow rates due to an LOOP, and the failure of transfer to house load. This leads to RCS and SG heatup and overpressure, to an early reactor trip (on low RCP speed) and to stabilization at HSD until the first operator action occurs.

When offsite power has been recovered, RHR is initiated and the plant is returned to a CSD condition.

Emergency Transient 2 - Long-Term Turbine Trip without TBS

This transient consists of: turbine trip without TBS, turbine trip resulting in RCS overpressurization, and turbine trip resulting in SS overpressurization. Initially at 100% FP, after stabilizing at HSD, the plant is placed in CSD.

Emergency Transient 3 - SG Tube Failure (One Tube) without Loss of Offsite Power

This transient is the double-ended rupture of a single steam generator U-tube resulting in a decrease in pressurizer level and reactor coolant pressure. This transient addresses partial SGTR and transients initiated in intermediate and CSD states.

Emergency Transient 4 - Small Secondary Side Break/Relief Valve Opening

A small secondary line break associated with emergency conditions is defined as a break equivalent to the accidental opening of a main steam safety valve (MSSV), an MSRT, or a TBS dump valve. The reactor is initially at HSD conditions which results in a more severe cooling transient. This transient also addresses small secondary side breaks at intermediate or CSD conditions.

Emergency Transient 5 - Faulty Opening of One PZR Safety Relief Valve

This event addresses opening of the PZR safety relief valve (PSRV) which induces primary side depressurization. This transient covers the spurious PSRV opening initiated at FP which bounds an occurrence at intermediate or CSD states.

This transient induces an RCS depressurization which leads to reactor trip, partial cooldown, and safety injection actuation on low PZR pressure signals. This transient is similar to Upset transient 5, but with a greater RCS depressurization gradient.

Emergency Transient 6 - RCS Pressurization between HSD and CSD

This event involves the mass addition and heat addition to the RCS during low temperature over pressurization (LTOP) conditions with the RCS pressurized. The initiating events include the inadvertent actuation of Safety Injection or CVCS charging which pressurizes the system due to mass addition and the start of an RCP with heat in the SG being transferred into the RCS.

3.9.1.1.4 Faulted Conditions

The following transients are considered faulted conditions:

Faulted Transient 1 - Primary Side Break (LBLOCA)

The largest primary side break belonging to faulted conditions is defined as a double ended guillotine RCS piping break which is considered an LBLOCA.

Faulted Transient 2 - Main Steam Line Break (MSLB)

The largest possible break in the steam lines is considered, which is the double ended guillotine break of the main steam line. This transient bounds all sizes of a MSLB and also envelopes a MSLB initiated in intermediate or shutdown states.

Faulted Transient 3 - MFW Line Break

The transient considered is a double ended rupture of the MFW line near the SG inlet. It covers all feedwater line break sizes.

If the break causes loss of MFW flow to an SG, then the SG is no longer able to effectively remove heat from the RCS (i.e., the MFW is not available for heat removal). This results in the initiation of the EFWS.

Faulted Transient 4 - Small Primary - Side Break (Small Break LOCA)

A small break loss-of-coolant accident (SBLOCA) under faulted conditions is defined as an RCS break with an equivalent diameter less than or equal to two inches, considered to be the diameter of small bore piping. The break is assumed to occur in the hot leg. After the break occurs, RCS pressure decreases and partial cooldown, safety injection, and reactor trip are actuated on low PZR level.

Faulted Transient 5 - RCP Locked Rotor

This transient results from a seizure of an RCP rotor along with an LOOP at the time of turbine trip. Flow through the faulted RCS loop decreases to zero, causing a reactor trip on a low-low loop flow rate signal. A turbine trip is initiated by the Protection System following a reactor trip and an LOOP is assumed to occur coincident with the turbine trip causing the remaining reactor coolant pumps to begin coasting down. For maximum primary and secondary pressure considerations, a failure of an MSRT to open is considered.

Faulted Transient 6 - Control Rod Ejection

This transient considers the inadvertent ejection of a control rod from the reactor vessel. The ejection of a control rod can be caused by a mechanical failure of a rod cluster control assembly (RCCA) drive mechanism casing displacing the RCCA vertically from the reactor core due to the high coolant pressure. This leads to a reactivity transient with high local energy deposition rates in the fuel and a pressure increase.

3.9.1.1.5 Testing Conditions

Hydrostatic tests for the US EPR are performed at various life cycles of the plant. A hydrostatic test transient is defined based on the ASME code and expected practice. ASME Section XI provides a schedule for specific times during the plant life cycle that require system pressure testing; however, allowances are provided to perform leakage tests in lieu of hydrostatic tests.

RCS components, including the secondary side of the steam generators, are subjected to individual hydrostatic tests prior to installation in the system. As part of the installed system hydrostatic tests, the RCS components, including the secondary side of the steam generators, are also subjected to hydrostatic tests after construction, but prior to initial plant startup. The total number of occurrences for the hydrostatic tests is applicable to each individual component in order to envelop both the individual component hydrostatic testing and the hydrostatic testing of components as part of the completed system.

ASME Section III provides requirements for hydrostatic tests that will be performed for the US EPR components prior to initial plant operation. Paragraphs NB/NC-6221 of ASME Section III specifies a minimum test pressure of 1.25 x design pressure for component and installed system hydrostatic tests. Hydrostatic tests performed prior to initial fuel loading will be performed at a temperature not lower than 70°F. In addition, for ASME Class 1 components, the test temperature shall be no lower than the reference temperature for nil ductility transition (RT_{ndt}) + 60°F as recommended in Nonmandatory Appendix G to ASME Section III, Division 1. Paragraphs NB/NC-6223 specifies a minimum hold time of 10 minutes; however, a hold time of one hour is expected due to the practical needs for test completion.

For any pressure test performed subsequent to initial plant startup, the requirements of ASME Section XI apply. ASME Section XI only requires a system leakage test, at nominal operating pressure and temperature, based on IWB/IWC-2500. The leakage test is typically performed at the end of a normal heatup cycle at hot shutdown conditions. This encompasses required pressure testing following repair or replacement activities in accordance with IWA-4540. The system leakage tests are not considered hydrostatic tests and, therefore, not included in the number of occurrences.

3.9.1.2 Computer Programs Used in Analyses

The following computer codes are used in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses of Seismic Category I components and supports. A complete list of programs will be included in the ASME Code design reports. The following information on computer codes is available for NRC inspection: author, source, version date, program description, extent and limitation of the program application, and code solutions to the test problems described in Appendix 3C and Appendix 3F.

- ANSYS and ANSYS CFX: ANSYS is a commercially available finite element analysis code for structural, stress, fatigue, and heat transfer analysis. It is used to perform stress and fatigue analyses of pressure vessels and their internals, as well as other complex geometries. Static and transient temperatures and pressures and applied mechanical loads can be modeled.

  ANSYS CFX is a commercially available finite element analysis code for computational fluid dynamic analysis. It is used in the analysis of the RPV internals to generate temperature profiles considering fluid heat transfer and internal heat generation (gamma heating).

  ANSYS and ANSYS CFX are owned and maintained by ANSYS, Inc. Validation of the ANSYS and ANSYS CFX computer codes is accomplished by executing verification cases and comparing the results to those provided by ANSYS, Inc. Each document that describes an ANSYS analysis includes information regarding the verification analysis and its results. Error notices from ANSYS, Inc. are processed and records pertaining to error notification, tracking, and disposition are available for NRC inspection.

- BWSPAN: Information on this computer code is provided in Appendix 3F.

- BWHIST, BWSPEC, COMPAR2, CRAFT2, P91232, EBDYNAMICS, CASS, and RESPECT: Information on these computer codes is provided in Appendix 3C.

- RELAP B&W: This is an advanced system analysis computer code designed to analyze a variety of thermal-hydraulic transients in light water reactor systems. As a system code, it provides simulation capabilities for the reactor primary coolant system, secondary system, feedwater trains, control systems, and core neutronics. Special component models include pumps, valves, heat structures, electric heaters, turbines, separators, and accumulators. Code applications include the full range of safety evaluation transients, loss of coolant accidents, and operating events. The code has been benchmarked to test facility data as documented in RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis (Reference 4).

- S-RELAP5: Information on this computer code is provided in Section 15.0.2. S-RELAP5 evolved from the AREVA ANF-RELAP code. S-RELAP5 was benchmarked against a series of LOFT experiments and against ANF-RELAP simulations.

- SUPERPIPE: Information on this computer code is provided in Appendix 3F.

- GTSTRUDL: Information on this computer code is provided in Appendix 3F.

- ROLAST: This is a one-dimensional best-estimate computer code that performs calculations of dynamic hydraulic loads for piping systems undergoing fast transients including water hammer phenomena. This AREVA developed code is used for single-phase flow. It can model behaviors of components such as pumps, valves, damped and undamped check valves, vessels with various boundary, and initial conditions. Typical code applications include operating and accidental fluid transient events in piping network systems such as check valve slam, rapid valve closure, pump start and stop, and pipe breaks. The code has been benchmarked to test facility data and plant data from existing nuclear power plants. Agreement between ROLAST calculations and test/measurement data has been obtained.

- S-TRAC: This AREVA developed computer code is based on the NRC Transient Reactor Analysis Code (TRAC Version-P) with hydraulic load calculations package added. TRAC-P is a thermal hydraulic analysis tool to calculate the transient reactor behavior of pressurized water reactors. S-TRAC features a one-, two- or three-dimensional treatment of the pressure vessel and its associated internals, a two-phase fluid non-equilibrium hydrodynamics model with a non-condensable gas field and solute tracking and a flow-regime-dependent constitutive-equation treatment. S-TRAC is used for two-phase fluid transients and multidimensional regions. Examples of code application include fluid transient loads in the reactor pressure vessel, steam generator (e.g., shroud, sparger, dryer, U-tubes), and the pressurizer relief piping system. The verification and validation of S-TRAC is based on the validation examples made for TRAC-P. Additionally, the code has been benchmarked to test facility data and plant data from existing nuclear power plants. Agreement between S-TRAC calculations and test/measurement data has been obtained.

- FatTool: FatTool is an AREVA developed MS Visual Basic program that is developed for the fatigue analysis of ASME Class 1 Piping. FatTool consists of two modules (i.e., the In-Air and Environmentally Assisted Fatigue (EAF) modules). The EAF module is a postprocessor of the In-Air module and incorporates the effect of Light-Water Reactor environment on the fatigue resistance of piping per the requirements of RG 1.207. The analysis methodology complies with the piping stress analysis requirements in ASME B&PV Code, Section III, Division I, NB-3600 and the EAF criteria and fatigue curves provided in NUREG/CR-6909 and RG 1.207. FatTool requires input of geometry parameters, forces and moments and thermal results from structural and thermal analyses codes like BWSPAN, P91232 and ANSYS. FatTool can perform a complete fatigue analysis (In-Air and EAF) using bending moments and thermal results from the external codes mentioned above. Alternatively it can also accept In-Air fatigue results from external codes and post-process them to evaluate the EAF results.

As addressed in Appendix 3F, there are three representative calculations from the analyses for the U.S. EPR design certification to be used in the benchmark program. These calculations utilize the piping analysis codes identified in Section 3F.5.1. As noted in Appendix 3F, pipe stress and support analysis will be performed by a COL applicant that references the U.S. EPR design certification. A COL applicant that references the U.S. EPR design certification will either use a piping analysis program based on the computer codes described in Section 3.9.1 and Appendix 3C or will implement a U.S. EPR benchmark program using models specifically selected for the U.S. EPR.

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Category I systems or components.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

Section 3.9.3 describes the analytical methods used to evaluate stresses for Seismic Category I systems and components subjected to faulted condition loading.

3.9.1.5 References

1. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Facility Components,” The American Society of Mechanical Engineers, 2004.

2. Deleted.

3. Deleted.

4. BAW-10164P-A, Revision 6, “RELAP5/MOD2-BAW – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses,” AREVA NP Inc., June 2007.

Table 3.9.1-1—Summary of Design Transients
Sheet 1 of 2

Category	Transient ID	Transient Description	Number of Occurrences
Normal	1	Plant Startup from Cold Shutdown to Full Load	240
	2	Plant Shutdown from Full Load to Cold Shutdown	205[1]
	3	Partial Heatup with Subsequent Shutdown	60
	4	Partial Shutdown with Subsequent Startup	60
	5	Power Ramp from Hot Shutdown to Full Load	3000
	6	Daily Load Follow	42,000[2]
	7	Frequency Control	1,500,000
	8	Unscheduled Power Variations	5250
	9	Unscheduled Fluctuations at Hot Shutdown	4000
	10	Partial Trip to 25 percent Full Power	560
Upset	1	Reactor Trip	90
	2	Turbine Trip	60
	3	Short-Term Loss of Power with Subsequent Return to FP	30
	4	Loss of Feedwater	60
	5	Spurious RCS Depressurization (Faulty Spraying)	15
	6	Reactor Trip with Excessive Secondary Side Heat Removal	15
	7	Excessive Feedwater Supply at Hot Shutdown	15
	8	Depressurization in the Secondary Side Leading to the Maximum Pressure Difference between the RCPB and the SSPB	15
	9	Unscheduled Pressure and Temperature Fluctuations between Hot and Cold Shutdown	4010
	10	Maximum SG Pressure with Open RCS	30
	11	Inadvertent Closure of One MSIV	15
Emergency	1	LOOP with Natural Circulation Cooldown	1
	2	Long-Term Turbine Trip without TBS Station	3
	3	SG Tube Failure (one tube)	1
	4	Small Secondary Side Break/Relief Valve Opening	1
	5	Faulty Opening of one PZR Safety Valve	1
	6	RCS Pressurization between Hot and Cold Shutdown	2

Table 3.9.1-1—Summary of Design Transients
Sheet 2 of 2

Category	Transient ID	Transient Description	Number of Occurrences
Faulted	1	Primary Side Break (LBLOCA)	1
	2	Main Steam Line Break	1
	3	MFW Line Break	1
	4	Small Primary - Side Break (Small Break LOCA)	1
	5	RCP Locked Rotor	1
	6	Control Rod Ejection	1
Testing	N/A	RCS Hydrostatic Tests	10

Notes:

1. Additional shutdowns to cold shutdown are included for the partial trip and reactor trip transients.

2. Although the U.S. EPR will be operated as a base-loaded plant, the reference U.S. EPR design provides robust features for the effects of load follow. Similarly, the structural design and analysis of the RCS, RCS components, RCS component internals, and systems ancillary to the RCS account for the effects of load follow.

3. The number, magnitude, and frequency of reversing dynamic seismic transient events corresponding to the Upset Level Service Condition to be used for fatigue analysis are evaluated per Note 12 of Table 3.9.3-1.

4. Stratification transients are not separate events. They occur because of particular combinations of flow and temperature and are an inherent part of several design transients. The contribution of normal and upset condition stratification cycles is considered in the fatigue analysis of piping systems. Note 8 to Table 3.9.3-1 addresses general and local thermal stresses resulting from system operating transients (i.e., pressure and thermal transients).

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