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 (SSCs) 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 Part 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 SSCs 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 Part 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 Part 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 Section 3.3 of U.S. EPR Piping Analysis and Pipe Support Design (Reference 2). 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.

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.

3.9.1.2 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. As noted in AREVA NP letter NRC:07:028 (Reference 3), 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 C and References 2 and 3.

- 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 Section 5.1 of Reference 2 and in Reference 3.
- BWHIST, BWSPEC, COMPAR2, CRAFT2, P91232, 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 NP 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 Section 5.1 of Reference 2 and in Reference 3.

As addressed in Reference 3, 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 5.1 of Reference 2. As noted in Reference 2, 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 an NRC-approved 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. Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), Request for Review and Approval of ANP-10264NP, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design," NRC:06:040, September 29, 2006.

3. Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Response to a Request for Additional Information Regarding AREVA NP Topical Report, ANP-10264(NP), 'U.S. EPR Piping Analysis and Support Design Topical Report,' (TAC No. MD3128)," NRC:07:028, July 13, 2007.

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

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

Category	Transient Description	Number of Occurrences
Faulted	Primary Side Break (LB LOCA)	1
	Main Steam Line Break	1
	MFW Line Break	1
	External Induced Transient	1
	RCP Locked Rotor	1
	Control Rod Ejection	1
Testing	Hydrostatic Test	3 for each component
	System Hydrostatic Test prior to Normal Operation	3
	Hydrostatic Test following Plant Operation	4

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 total number of strong stress cycles for emergency events is less than 25.